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Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a Westinghouse PWR

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ABSTRACT

Pacific Northwest Laboratory (PNL) performed a probabilistic risk assessment to develop estimates of core-melt frequency and public risk due to control system failures in a Westinghouse pressurized water reactor. Value/impact analyses of proposed systems modifications to prevent the control system failures were also performed. Four control system failure modes were analyzed: 1) overfill, 2) overcool, 3) overpressure, and 4) steam generator tube rupture. For each mode, two failure sequences were postulated. These analyses were based on the results of failure modes and effects analyses previously performed at Idaho National Engineering Laboratory and conducted in support of the U.S. Nuclear Regulatory Commission's program for Unresolved Safety Issue A-47: Safety Implications of Control Systems.

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EXECUTIVE SUMMARY

Pacific Northwest Laboratory (PNL) has performed a probabilistic risk assessment (PRA) of control related failures in light water reactors for the U.S. Nuclear Regulatory Commission (NRC). This work was performed in support of the NRC's Unresolved Safety Issue (USI) A-47 program: Safety Implications of Control Systems. This report specifically focuses on control failures in a representative Westinghouse pressurized water reactor (PWR). The PRA was based on failure modes and frequencies developed for the PWR by Idaho National Engineering Laboratory (INEL).

In addition, PNL has performed a value/impact analysis of proposed resolutions to correct control deficiencies identified by the A-47 program. Value/impact analyses are required by the NRC as input into the regulatory decision making process to insure that the need for and consequences of cost-effective regulatory actions are identified (U.S. NRC 1983a). Cost/benefit analyses are not the sole or even principal basis for decisions, but they do provide one consideration for the decision making process. The purpose here was to provide a screening tool of potential resolutions only.

GENERAL OVERVIEW

The report addresses the following major topics:

- a listing of control failures identified as being of concern to the A-47 program
- discussion of the safety implications of failures and progression to core-melt scenarios
- calculation of risk
- introduction of resolutions to mitigate or eliminate failures
- estimates of potential risk reduction with implementation of proposed resolutions
- cost of resolutions
- presentation of resulting value/impact ratios.

These topics will be summarized briefly in this section, which is followed by a more detailed summary of the technical analysis and results.

Control Failures of Interest

The A-47 program focused only on those control failures that could initiate a plant response more severe than previously analyzed in design basis accidents, or failures that could cause plant conditions to exceed operating technical

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specifications. Using the H. B. Robinson plant as a representative Westinghouse PWR, INEL identified four general failure modes dealing with overfill and overcool of steam generators, overpressure of the reactor vessel, and steam tube ruptures coupled with aggravating control failures. Two specific control failures were identified for each failure mode. PNL then examined the identified failures for safety implications.

With the exception of loss of offsite power, the scope of the INEL investigation did not include the consideration of events external to the plant (e.g., seismic events, aircraft crashes, etc.). Such events are thus not considered in this report.

Safety Implications

Overfill of the steam generator was initiated by failures in the main feedwater control and high steam generator water level trip circuits. This has the potential to lead to water pouring into the steam lines if not terminated by the operator, possibly resulting in steam line damage including major steam line failure. A large uncertainty currently exists concerning this potential, so a high probability of main steam line break (MSLB) given spillover of water into the steam lines was assumed. The potential for MSLB leading to severe core damage was then modeled using the results of the Oak Ridge National Laboratory (Minarick and Kukielka 1982) and the Institute for Nuclear Power Operations (INPO 1982) in investigating precursors to severe core damage. In addition, the potental for MSLB progressing to steam generator tube rupture (SGTR) and core-melt was modeled using the result of the NRC evaluation of steam generator tube integrity (U.S. NRC 1985).

Overcool scenarios were initiated by control failures lifting relief valves in the steam lines. Risk was modeled by again considering the valve lift as a steam line break, and considering the potential for inducing an SGTR given rapid depressurization of the steam line. The potential for pressurized thermal shock (PTS) leading to vessel failure was also considered.

Overpressure scenarios were caused by letdown valve failures blocking discharge flow from the reactor vessel, or inadvertent high pressure injection to the vessel. The potential for PTS was considered.

Steam generator tube rupture (SGTR) scenarios with aggravating control failures were modeled as loss of coolant accidents (LOCAs), which result in loss of the primary coolant inventory in a PWR.

Risk

Event trees were developed for the above scenarios, estimating the probability of system failures needed to terminate the progression of the accident to core damage or core-melt. This represented a conditional probability of core damage given the initiating event. When multiplied by the initiating event frequency estimated by INEL, the frequency of core damage and core-melt was obtained. INEL median estimates of initiating frequency were used, with a PNL best engineering estimate of the probability of subsequent protective system failure. The highest core-melt frequency predicted was for

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an overcool scenario involving the inadvertent lifting of a power operated steam relief valve (PORV), with an estimated core-melt frequency of 8.3E-07/py. All other scenarios had estimated core-melt frequencies over one order of magnitude less, primarily due to the extremely low initiating frequencies predicted by INEL.

The resulting core damage and core-melt scenarios were then compared to similar scenarios in reactor PRAs, and the scenarios were assigned to representative radionuclide release categories. The man-rem exposures associated with these release categories were developed in the Value/Impact Handbook (Heaberlin et al. 1983). The net result was an estimated man-rem/py of public exposure associated with the postulated failures. Again, the highest estimated public risk was associated with the PORV overcool scenario, with risk put at 3.9 man-rem/py. All other risks were again at least one order of magnitude less.

When put in perspective with the risk estimated for other nuclear safety issues (U.S. NRC 1983b), the above estimate of core-melt frequency and risk is extremely small. When the NRC methodology is applied for prioritization of efforts to resolve safety issues, it becomes apparent that no fixes can be justified for this plant on a cost/benefit basis.

Risk Reduction/Cost/Value Impact

The report also develops the remaining topics identified above:

- risk reduction with modifications
- costs of modifications
- resulting value/impact ratios.

However, these subjects are not discussed in this general overview, because of the conclusion that the control system failures identified result in too low a core-melt frequency and associated risk to warrant plant modifications from a value/impact perspective.

TECHNICAL OVERVIEW

In the text below, a more in-depth technical summary of this report will be given. The potential control system failures identified by INEL again fall into four main scenarios: 1) overfill, 2) overcool, 3) overpressure, and 4) steam generator tube rupture (SGTR). For the first three scenarios, no failure is postulated by INEL that initially impacts the success of the core cooling function. In fact, the overfill and overcool scenarios are just the opposite, with excessive cooling the initial problem. The scenarios were postulated so that cooling rates exceed standard technical specifications for the plant. The overpressure scenario likewise creates conditions that exceed technical specifications.

In terms of public risk, PNL assumed that the potential for a transition to an undercooling event that could affect core integrity was the common connection among all these scenarios. The only mechanisms identified in this respect are the potential for inducing a main steam line break (MSLB) or a steam generator

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tube rupture (SGTR) that would, in turn, impact the core cooling function. The SGTR scenarios already represent a primary-side LOCA. In that case, INEL postulated additional control system failures to further aggravate the rupture event.

The events described above, if allowed to progress to an MSLB could also represent cooldown transients that require plant shutdown. Steam line breaks or ruptures on the secondary side of a PWR were considered cooldown transients in the WASH-1400 report; however, it was determined that this was not a credible pathway to core-melt. It was recognized that fuel damage could occur with release of fission gases in the fuel/cladding gap, but no credible fuel melting scenarios were identified.

However, MSLBs in the Oak Ridge and INPO studies (INPO 1982, and Minarick and Kukielka 1982, respectively) were considered as precursors to severe core damage. To be conservative in this analysis, the MSLBs were then modeled as developed in the ORNL study and as updated by INPO. In addition, the potential for inducing SGTR given MSLB was also developed for each sequence when appropriate. The SGTR represents a LOCA which can then progress to core-melt.

More recent considerations of SGTR events, however, have significantly increased the potential for core-melt if the primary coolant system cannot be successfully depressurized before exhaustion of injection water supplies. This type of accident differs from small break LOCAs (SBLOCAs), where water would usually be available from building sumps for recirculation. Further, the potential for progression to core-melt given an SGTR is several orders of magnitude greater for scenarios in which the main steam isolation valves (MSIVs) cannot be used to isolate the affected steam generator. This would be true for pipe breaks or valve lifts above the MSIVs.

Given the uncertainty in the potential for MSLB and where it may occur, PNL conservatively used a high probability of MSLB above the MSIVs, making isolation of the affected steam generator impossible and successful reactor coolant system (RCS) depressurization unlikely. The same scenario was used for inadvertent power operated relief valve (PORV) lifts and failure to isolate with block valves. Failure of recirculation water supplies would also result in the failure of containment sprays; thus PNL assumed that such scenarios would be associated with a severe WASH-1400 release category 2. The net result was to predict a dominant contribution to overall core-melt and risk from progression of overfill or overcool control system failures to SGTR. However, if the A-47 program generates more realistic evaluations on steam line dynamics and the potential for failure as a function of location (i.e., above or below the MSIVs), the contribution to core-melt and risk from assuming progression of the control system failure to SGTR could easily be reduced by several orders of magnitude.

The impact of external events may potentially be of interest in the A-47 program. The conditional probability that control system failures resulting from an external (e.g., seismic) event may be higher in nonsafety grade equipment than in safety grade equipment would be the primary concern. Control failures could then possibly aggravate systems and operator response to the event. However, the role of external events on control system failure modes or

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rates was not investigated by INEL in their definition of failure scenarios. Since PNL's analysis is based on the scenarios defined by INEL, external events were excluded from the PNL analysis of the safety significance of the scenarios identified by INEL.

Pressurized Thermal Shock

The potential for pressurized thermal shock (PTS) and vessel rupture was also examined to the extent possible at this time. Results from the PTS program for severe cooldowns due to steam line break or valve lift and HPI pressurization of the vessel are thought to bound the types of failures predicted in the A-47 program that may produce PTS-type events. Preliminary calculations by ORNL for such scenarios give conditional probabilities of vessel failure of 1E-04, using 270°F as a conservative reference temperature for nil ductility transition. Given a vessel failure probability of 1E-04, the overall frequencies of some control failures progressing to PTS, vessel failure and core-melt will be 20 percent of the core-melt frequency for progression to MSLB, SGTR, and core-melt. The PTS contribution would thus become of interest but would not significantly change the overall estimate of core-melt and public risk and value/impact conclusions. Overpressure scenarios likewise gave a low core-melt frequency due to PTS-induced vessel failure.

If the actual transition temperature (130°F) for the H. B. Robinson plant is used, the conditional probability of vessel failure for the representative PTS overcool scenario drops by several orders of magnitude. The contribution of PTS and vessel failure to core-melt then becomes insignificant for A-47 events. The PTS program concludes that plants with such low transition temperatures present an acceptably low risk with respect to PTS induced vessel ruptures.

The latter estimate is thought to be the best measure of risk due to PTS induced failures in the A-47 program. It is appropriate for conservative calculations to be used within the PTS program for evaluation and resolution of the PTS issue. However, these conservative assumptions should not be transferred back into related issues if undue weighting or influence of specific safety concerns is to be avoided in a calculation of relative risk. This best estimate then indicates that PTS plays a minimal role in core-melt and risk for the A-47 analysis of the Westinghouse H. B. Robinson PWR at this time.

The results of the analysis of PTS potential are summarized in Table S.1. Note that all estimated core-melt frequencies are quite low. Again, no distinction was made between core damage and core-melt. This is considered conservative for steam line break scenarios with the frequency of core damage scenarios more likely to be reduced by an order of magnitude to represent core-melt. Several sequences resulted in calculated core-melt frequencies considerably below a significant level. These are reported in the summary tables as less than 1E-10/py.

The MSLB contribution for overfill and overcool scenarios shown in Table S.1 is the product of the initiating frequency, the probability of operator failure to terminate the event if possible, the estimated probability of an

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TABLE S.1. Summary of the INEL and PNL Estimates of Accident Initiator Frequencies, Core-Melt Frequencies and Public Risk

Sequence Initiator	INEL Accident Initiating Frequency median (/py)	PNL Core-Melt Frequency Best estimate (/py)	Public Risk Best estimate (man-rem/py)
Steam Generator Overfill Sequence 1	1.4E-03		
Transient Shutdown		2.8E-09	1.5E-02
MSLB		7.7E-10	2.1E-03
SGTR		5.8E-08	2.8E-01
Subtotal		6.2E-08	3.0E-01
Steam Generator Overfill Sequence 2	5.4E-08		
Transient Shutdown		<1E-10	<1E-04
MSLB		<1E-10	<1E-04
SGTR		<1E-10	<1E-04
Subtotal		<1E-10	<1E-04
Reactor Coolant System Overcool Sequence 1	2.6E-07		
MSLB		<1E-10	<1E-04
SGTR		<1E-10	<1E-04
Subtotal		<1E-10	<1E-04
Reactor Coolant System Overcool Sequence 2	1.8E-02		
MSLB		1.1E-08	2.9E-02
SGTR		8.2E-07	3.9E+0
Subtotal		8.3E-07	3.9E+0
Reactor Coolant System Overpressure Sequence 1	1.5E-07	<1E-10	<1E-04
Reactor Coolant System Overpressure Sequence 2	3.7E-04	<1E-10	<1E-04
Steam Generator Tube Rupture Sequence 1	2.0E-03 7.0E-06 with SGTR	1.3E-08	6.9E-02
Steam Generator Tube Rupture Sequence 2	3.2E-03 1.1E-05 with SGTR	<1E-10	<1E-04
TOTAL		9.1E-07	4.3E+0

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MSLB, and the estimated probability of core damage given MSLB. The operator error term was established at 0.1 in most cases, given the condition of the plant and positive instrumentation readings available. The probability of MSLB for the dominant event tree pathways was then established at 0.5, down from the 1.0 used in the BWR analysis[a] due to the lower temperature and pressure levels involved for overfill or overcool. The net effect was to reduce the INEL initiating frequency for MSLB by a factor of approximately 0.05. For sequences involving steam-side PORV lift and sticking, which create an MSLB directly, the probability of operator failure to close a block valve or block valve failure was set at 0.055, based on consideration of PORV/block valve reliability in Safety Issue 70.

Finally, the probability of core damage given MSLB was set at 1.1E-05, based on event trees from the ORNL/INPO studies (Austin et al. 1985; INPO 1982). These considerations effectively reduced the estimated frequency of inducing core damage to 1.0E-08/py. The accepted conversion to core-melt further reduces this probability by over an order of magnitude. However, this was conservatively equated to core-melt for simplicity, assigning the frequency to PWR release categories 3, 5, and 7 with the probability of 0.5, 0.0073, and 0.5, respectively. This is representative of transient-initiated core-melt sequences, with ultimate failure of the long-term core cooling function. The predicted public dose was then less than 1 man-rem/py for all MSLB sequences.

For overfill/overcool scenarios postulated by PNL to progress to SGTR, the probability of inducing an SGTR given an MSLB was set at 0.034, based on considerations of such events for the NRC Steam Generator Tube Integrity Program. The conditional probability of progression to core-melt, given the SGTR, was then calculated to be 2.44E-02, dominated by the assumptions outlined above for isolation of the affected steam generator.

For those sequences involving a PORV or feedwater under operator control, the uncertainty in correct operator action has been discussed in the analysis. Only Overfill Sequence 2 provides the operator with an unambiguous indication of changing steam generator levels. Thus the role of the operator in diagnosing and terminating the scenarios introduces some uncertainties. The analyses tried to treat these uncertainties conservatively. Note that the role of the operator will also impact any potential "fixes" of the system by introducing operator training or control-room, human-factors-engineering considerations.

The overfill scenarios were assumed to lead to steam line break to provide any sort of core-melt sequence initiator. The basic uncertainty in the potential for inducing a steam line break still exists in the PWR analysis, as it did in the BWR analysis. However, the overfill analyses in this case did not actually progress to spillover of water into the steam lines. Power levels were

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(a) W. E. Bickford and A. S. Tabatabai. 1985. Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a General Electric Boiling Water Reactor. PNL-5545, unpublished draft, Pacific Northwest Laboratory, Richland, Washington.

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also low or at startup for the PWR. The potential for water hammer and steam line break was adjusted accordingly (compared to the BWR analysis), giving again what is thought to be a conservative assumption for steam line break.

INEL postulated two SGTR scenarios. However, no control system failures were identified as initiators of the tube rupture event. These sequences assumed control system failures independent of the tube ruptures as contributions to the LOCA, providing an insignificant contribution to core damage potential. This is primarily due to the already extremely low frequency calculated by INEL for the initiating events of interest. Both SGTR scenarios are accompanied by independent control system failures used to increase the severity of the transient, resulting in the low probability. Note, however, that the necessary system response is still that required to prevent core damage for a simple steam generator tube rupture.

The total predicted core-melt frequency is then 9.1E-07/py for all sequences considered, with a public dose of 4.3 man-rem/py. This is considered to be a nondominant contribution to the overall core-melt frequency of a PWR. For the WASH-1400 Surry Westinghouse PWR, the overall core-melt frequency was approximately 6E-05/py, with the contribution from SBLOCAs contributing approximately 2.9E-05/py. Because the LOCAs play a dominant role in PWR risk, it is assumed that the risk calculated here will also represent only several percent of total overall plant risk.

Value/Impact Analysis

To analyze the value/impact associated with this issue, it was necessary to postulate possible design changes to alleviate the control system failures identified by INEL. For steam generator overfill, the failures are similar to those postulated by INEL for initiating overfill in a BWR (i.e., instrument line leakage, level transmitter failure)(Bruske et al. 1985). Those portions of the value/impact analysis then refer to the previous PNL examination of overfill in a GE BWR.

In addition, as with the BWR, it is expected that there is significant interaction of the operator with control systems. The operator's role was conservatively estimated in the core-melt calculations presented above. However, no improvement in performance was postulated as a remedy to failures identified by INEL. With control of vessel water levels the focus of training and procedures upgrades since the Three Mile Island (TMI) accident, a significant reduction in operator error for main or auxiliary feedwater failures, PORV lifts, etc., could reasonably be expected to reduce the chance of progression of simple control failures to more serious accidents.

Before requiring major equipment modifications, it is thought that the analysis of USI A-47 should recognize this important role of the operator and make appropriate recommendations. This could include a more detailed examination of the time available to the operator, signals and indications available, and current procedures. Recommendations could also be made concerning other specific task action items established specifically to deal with operator actions during transients. These items are better geared to deal with the potential for reducing operator error in general and would insure a consistent approach to operator interactions.

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The resolutions postulated were directed at reducing the rate of control equipment failures identified by INEL. These are shown in Table S.2, along with the estimate for reduction in core-melt frequency, public risk reduction, cost, and resulting value/impact. The addition of another level transmitter (LT) in a 2-out-of-4 trip logic, which was thought to be the most cost-effective measure to counteract the control system failures identified by INEL for feedwater overfill in the BWR, has a slightly less favorable value/impact for the PWR. This is because the sequence of interest stabilized in the PWR, allowing more time for operator action. For this reason, the failure probability for this scenario was reduced to 0.1. This scenario is also less significant in PWRs because of the reduced potential for steam side problems progressing to core damage.

In the PWR, the overfill sequence driven by auxiliary feedwater (AFW) overfill could be terminated by the simple addition of a high-level AFW trip. The steam side break caused by PORV lift could likewise be eliminated through the use of block valves. Note that both of these fixes are already under operator control. Fixes to reduce the frequency of such events could focus on operator training and procedures, or on providing automatic actuation of shutoffs or block valves as assumed here. A more detailed examination of current procedures and time/signals available to the operator would be required to determine which modification is more appropriate.

The value/impact ratios calculated for the above fixes do not approach the 1 man-rem/$1000 figure of merit, and thus do not appear beneficial at this time. As noted above, further information on the potential for MSLB and non-isolatable SGTR could easily further reduce the estimated core-melt and public risk by several orders of magnitude. The value/impact ratios must be considered in light of this uncertainty.

Note that in the absence of a more rigorous analysis, the upper bounds on the core-melt frequency and public risk for all of the above scenarios are approximately a factor of 10 greater than the median estimates. The costs likewise represent a best engineering estimate. Thus a certain amount of latitude is needed in interpreting the value/impact ratio. However, the development of these accident initiators to core-melt is thought to reflect a conservative approach to estimating the impact of these failures on plant engineered safety systems. Cost estimates likewise tend to underestimate the true cost of nuclear plant modifications. These factors, when combined, tend to further reduce the estimated value/impact ratios as given above.

Possible plant modifications to reduce the frequency of overfill can be bounded by comparison to the proposed Safety Goal cost/benefit guideline of $1000/man-rem averted. Assuming a 30-year effective plant life, the total possible risk reduction is approximately 723 man-rem/reactor. If the costs of potential corrective features are compared to the benefits on the basis of $1000/man-rem averted, then an upper bound of approximately $723,000 can be placed on the costs of corrective features. Note again, however, that highly conservative assumptions were made concerning the probability of MSLB and subsequent SGTR given overfill. Further, assuming that breaks could occur above the MSIVs, making isolation of affected steam generators impossible, the resulting probability of core-melt is high. More detailed information on the potential or MSLB and possible break locations could thus reduce the estimated contribution to core-melt and public risk by several orders of magnitude.

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TABLE S.2. Summary of the Value-Impact Analysis of the Proposed Resolutions

Proposed Resolution	Scenarios Affected	Estimated Cost ($)	Estimated Risk Reduction (man-rem)[a]	Value/ Impact Ratio (man-rem/ $1000)[a]
Better Weld Integrity	SG Overfill Sequences 1 & 2	$1.13E+05	1.0E-03	<1E-04
Hardened Instrumentation Lines	SG Overfill Sequences 1 & 2	$3.22E+04	7.8E-03	2.4E-04
Automatic Shutoff of the Auxiliary Feedwater	SG Overfill Sequence 1	$2.69E+04	9.0E+0	0.3
New Level Transmitter with 2-out-of-4 Trip Logic	SG Overfill Sequence 2	$1.50E+05	7.8E-04	<1E-4
Automatic Actuation of Isolation Block Valves	RCS Overcool Sequences 1 & 2	$1.23E+05	20.5	0.17
Modifications to Valve Controller Logic	RCS Overcool Sequences 1 & 2	$1.23E+05	20.8	0.17
Independent Power Source to the Letdown Valve and PORVs	RCS Over-Pressure Sequence 1	$6.40E+04	negligible	negligible
Modifications of LTOP Mode Switch	RCS Over-Pressure Sequence 1	$1.00E+04	negligible	negligible
Modifications to the PORV Control	RCS Over-Pressure Sequence 1	$3.20E+04	negligible	negligible
Logic Circuit Modification	RCS Over-Pressure Sequence 2	$8.00E+04	negligible	negligible

(a) "Negligible" indicates scenarios with core-melt frequencies less than 1E-10/py and resulting value/impact ratios less than 1E-04.

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The calculations in this report are intended primarily to provide perspective on the risks and costs of the A-47 issue. The NRC has established the safety goals for evaluation during a 2-year period, but not for regulatory use during that period. Furthermore, the proposed benefit-cost guideline, even if adopted, would not be the sole or even the principal basis for decisions on safety improvements; rather, it would be one consideration in such decisions. This report presents only the preliminary analysis of the costs and benefits associated with possible design features to correct control system failures. It is suggested that a more detailed analysis into the possible negative impacts on control system performance be required before modifications postulated in this report are implemented in existing nuclear plants.

Areas of Likely Conservatism

A best engineering estimate of failure probabilities was used whenever possible in the analysis of core-melt and risk for the control failures identified. Some uncertainty does exist, however, in several factors, with the analysis carried through using what is thought to be a high failure probability. This in turn weights the estimated core-melt frequency and public risk to higher values. These include the following:

- Operator Error: The probability assumed for failure of the operator to diagnose and terminate the scenarios ranged from 0.5 for scenarios with misleading or conflicting information or rapid progression (i.e., overfill in several minutes) to 0.1 for scenarios with non-conflicting information and alarms. Actual operator response may be better than estimated, particularly in plants with simulator programs stressing proper diagnosis of failures.

- Steam Line Break: MSLBs in PWRs were not assumed to be associated with core-melt in the WASH-1400 study. More recent studies have equated MSLB with core damage equal to or less severe than a core-melt in terms of radionuclide release by up to a factor of 30. PNL's study equated the consequences of MSLB with core-melt.

  The probability of MSLB given spillover into the steamlines at power was assumed to be 1.0, decreasing to 0.5 for spillover after shutdown. Although several spillover events in U.S. commercial plants have resulted in support damage, no steam line failures have occurred. Breaks were assumed to occur with a 50 percent probability above the MSIVs, making isolation impossible. The MSLB was further assumed to have a significant probability of inducing a steam generator tube rupture (SGTR), with the combination of SGTR and unisolatable MSLB leading with high probability to core-melt in a PWR. This high probability of failure to recover is due primarily to depletion of the reactor water storage tank (RWST) water supply before depressurization of the reactor can be achieved. This gives no credit to other operator-initiated means of maintaining a water supply.

  Further information on the probability of break for various overfill scenarios and the break location could significantly reduce the risk associated with these scenarios, as would a more realistic analysis of operator initiated actions to restore water supplies and avoid core-melt.

- Transient Shutdown: The initiating event would cause a transient-induced plant shutdown, with loss of the power conversion system (PCS) representing a serious precursor to core-melt in PWRs. A high probability of loss of the PCS given spillover was assumed here, but contributed insignificantly to risk in this analysis due to the low initiating frequency.

- Release Categories: The WASH-1400 release categories most representative of the core-melt scenarios in this analysis were used to estimate risk, with the risk per event taken from the Value-Impact Handbook (Heaberlin et al. 1983). Ongoing evaluations of the source terms for various core-melt scenarios indicate that the WASH-1400 release categories may overestimate risk by up to several orders of magnitude, resulting in a lower risk being attributed to each scenario.

- Costs: Estimates of the costs associated with modifications in nuclear plants typically underestimate the final costs, even when accompanied by an extensive engineering cost study. Higher than expected costs would further lower the value/impact ratios estimated here for proposed modifications.

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1.0 INTRODUCTION

This report examines the probability of core-melt and risk associated with control system failures in Westinghouse pressurized water reactors (PWRs). This work is based on the control system failures and failure frequencies identified by Idaho National Engineering Laboratory (INEL) (Ransom et al. 1985).

The Carolina Power and Light H. B. Robinson Unit 2 nuclear plant was used as the reference design in the INEL investigation because it is a three-loop PWR design. The INEL report identified eight initiating events which had a potential to cause severe core damage. These are shown in Table 1.1.

TABLE 1.1. INEL Identified Control System Accident Initiators for Westinghouse PWRs

1. Failures that result in increased feedwater flow rates which subsequently lead to the auxiliary feedwater flow causing a steam generator overfill.

2. Failures that result in excessive feedwater flow rates with subsequent failure of the steam generator high water level trips.

3. Failures that result in inadvertent steam dump operation with the reactor at power.

4. Failures that result in inadvertent opening of the steam line relief valves with the reactor plant in hot shutdown (average temperatures less than 547°F).

5. Failures that result in loss of letdown flow and pressure relief capabilities with the reactor plant in cold shutdown.

6. Failures that result in inadvertent safety injection (SI) initiation with the reactor plant being heated from cold shutdown with the pressurizer power operated relief valves (PORVs) set for normal full power operation.

7. Failures that result in steam line safety or relief valves failing open and in high feedwater flow rates concurrent with a steam generator tube rupture on the affected steam generator.

8. Failures that result in steam line safety or relief valves failing open and in high feedwater flow rates concurrent with a steam generator tube rupture.

The failure mechanisms causing the above systems failure were identified and sequence probabilities were calculated by the INEL. These probabilities are given in Table 1.2. The description of the sequences has been abbreviated to the titles used by INEL.

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TABLE 1.2. INEL Estimate of Accident Initiator Frequencies

Sequence	Median Value Per Reactor Year	90th Percentile Per Reactor Year
1. Steam Generator Overfill Sequence 1	1.4E-03	5.5E-03
2. Steam Generator Overfill Sequence 2	5.4E-08	5.5E-07
3. Reactor Coolant System Overcool Sequence 1	2.6E-07	1.4E-06
4. Reactor Coolant System Overcool Sequence 2	1.8E-02	5.0E-02
5. Reactor Coolant System Overpressure Sequence 1	1.5E-07	7.7E-06
6. Reactor Coolant System Overpressure Sequence 2	3.7E-094	1.2E-03
7. Steam Generator Tube Rupture Sequence 1	2.0E-03(a)	1.5E-02(a)
8. Steam Generator Tube Rupture Sequence 2	3.2E-03(a)	1.9E-02(a)

(a) For steam generator tube rupture events, the probabilities are shown for the aggravating failures only. Probabilities for coincident tube rupture will be added in this report in later chapters.

The purpose of this report is to analyze the accident progression and determine its effect on core-melt and public risk. The accident initiators will be examined in the following chapters, and event trees developed to determine core-melt sequences. The performance of the relevant safety systems given the assumed initiating conditions will then be examined to determine any impact on system performance. The probability of sequences leading to core-melt will then be estimated. Finally, the likely release categories associated with the sequences will be determined and an estimate of public risk associated with the accident will be made. An attempt will also be made to put these risks into perspective through a comparison to similar types of transient-induced sequences and to overall plant risk in a comparable Westinghouse PWR. Lacking a probabilistic risk assessment (PRA) of the reference H. B. Robinson plant, generic event trees will be developed to consider accident progression of the initiating events towards core damage and core-melt.

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In addition, plant modifications are postulated to eliminate or reduce the frequency of the above failures. Costs associated with implementing these fixes will be estimated, and a value/impact ratio presented in terms of man-rem of potential public exposure reduced per $1000 spent in the modifications.

The purpose of this work is to provide a screening tool to determine the magnitude of safety concerns and the value associated with possible resolutions. This insures that the need for and consequences of cost-effective regulatory actions are identified. Cost/benefit analyses are not the sole or even principal basis for decisions, but they do provide one consideration for the decision making process (U.S. NRC 1983a).

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2.0 STEAM GENERATOR OVERFILL SEQUENCE 1

This sequence involves failures in the feedwater control that result in increased feedwater flow rates, subsequently leading to the auxiliary feedwater flow causing a steam generator overfill.

2.1 INITIAL PLANT CONDITIONS FOR STEAM GENERATOR OVERFILL SEQUENCE 1

The initial plant conditions assumed by INEL for this sequence set the plant at 5 percent reactor power with the rod control, feedwater control, and turbine electrohydraulic control in manual, with all other systems in automatic. The low initial power was used in the INEL scenario to get the largest possible steam flow/feedwater flow mismatch.

A failure in the feedwater level or control system, causing an increased feedwater flow, was then assumed. Calculations were based on a 10%/s increase in the feedwater flow. The main pumps were then tripped on the high level trip signal at 36 seconds into the transient, with the overfill continuing with the auxiliary feedwater pumps.

The INEL calculations indicate that water carry-over can be inferred from the steam generator dome steam qualities, with carry-over predicted around 205 seconds into the transient. Significant carry-over then would occur in slightly over 3 minutes after main feedwater trip. Note that power levels would increase in response to the feedwater increase. INEL noted that for initial reactor power levels above the 5 percent point, the power increase would be sufficient to preclude steam generator overfill due to the parallel increase in the steaming rate.

The sequence of events as postulated by INEL and the required time for each event to take place due to the feedwater failure are shown in Table 2.1.

TABLE 2.1. Sequence of Events for Steam Generator Overfill Sequence 1

Time (s)	Event
0.0	Transient initiated by opening MFW control valve
10.0	MFW valve wide open
22.0	Steam dump valve flow peaked at 39 kg/s
35.7	MFW pump tripped on 75 percent SGA NR level
36.1	SGA steam line check valve closed
55.0	Steam dump valve closed
127.0	Steam dump valve opened to control steam header pressure

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TABLE 2.1. (Cont'd)

Time (s)	Event
150.0	SGA boiler volumes at saturation pressure and begin voiding
196.0	SGA steam line check valve reopened
210.0	SGA NR level reached 96.6 percent
240.0	Transient terminated

2.2 ACCIDENT PROGRESSION ANALYSIS FOR STEAM GENERATOR OVERFILL SEQUENCE 1

A review of the INEL accident sequence for steam generator overfill indicates that there is initially no failure of the primary containment or cooling system postulated. For the accident to impact the integrity of the core, it must progress to an undercooling scenario.

For this accident scenario the principal hazard is believed to be an induced steam line break on the secondary side with the potential for causing a steam generator tube rupture (SGTR). Figure 2.1 presents the event tree leading up to possible steam line break and SGTR using logic similar to that developed for analysis of overfill in the BWR. The steps in this event tree are discussed below.

Initiating Event. The initiating event for this scenario is the feedwater control opening the main feedwater valve, followed by feedwater trip and continued overfill with the auxiliary feedwater. The median value for this is placed by INEL at 1.4E-03/py, with an upper bound of 5.5E-03/py. For the purposes of this examination, the INEL estimate of 1.4E-03/py will be used.

Figure 2.1 then shows the initiating event, with overfill continuing following trip of the main feedwater turbines on high level. The auxiliary feedwater pumps, being electrically driven, are not impacted by the degrading steam quality in the steam lines. Operator action is required to terminate the overfill.

Operator Isolates Auxiliary Feedwater. The transient can be terminated by the operator simply isolating the auxiliary feedwater flow. Given that the operator is trained to monitor the feedwater flow and water level during startup, the probability of the operator detecting the overfill condition and correcting it is considered to be very likely. This is further increased by the high water level indication, followed by alarm and trip of the main feedwater system. The probability of correct action was put at approximately 0.5 in the analysis of overfill for BWRs, but given the main feedwater trip and uniform level readings (i.e., no instrumentation failures), the probability of operator failure to terminate the auxiliary feedwater flow will be put at 0.1 for this examination (i.e., successful termination is thought to be more likely).

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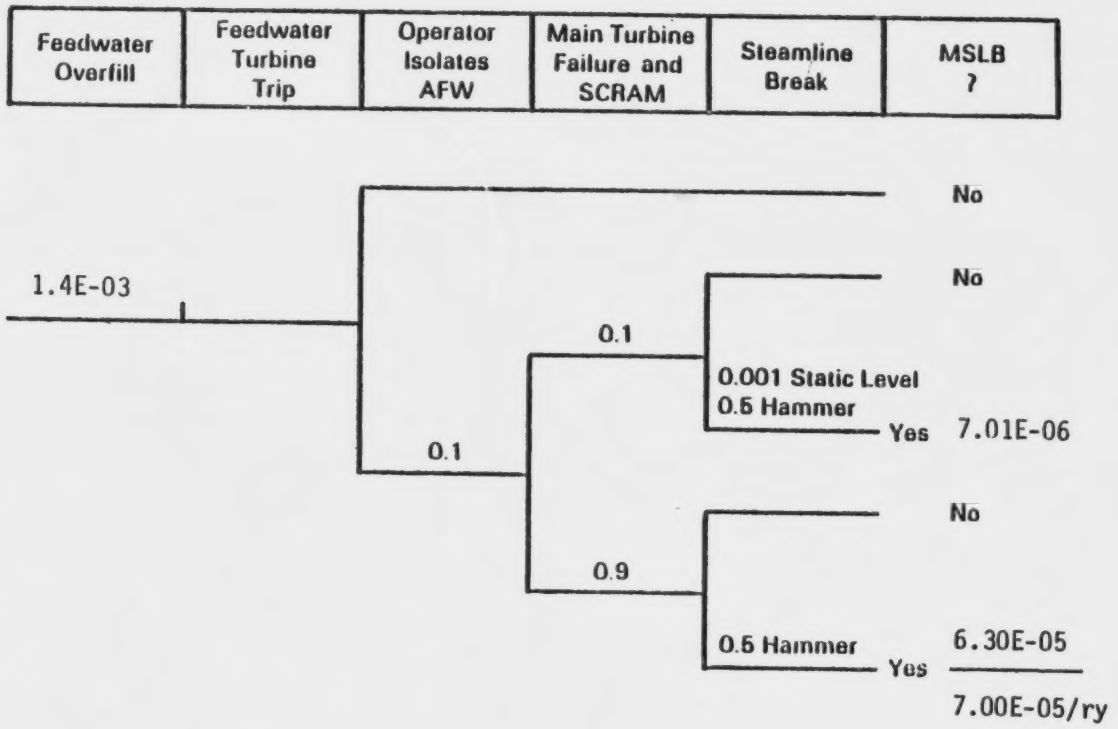

FIGURE 2.1. Feedwater Overfill Sequence 1: Progression to Steam Line Break

Main Turbine Failure and SCRAM, and Steam Line Break In the BWR analysis, degrading steam quality and excessive moisture into the steam lines was identified as a possible failure mechanism for the main turbine, causing SCRAM. The potential for steam line break was considered worse if spillover of water occurred during actual operation with a significant steam mass flow versus a degrading steam flow or static steam in the main lines. If spillover occurred during operation, a probability of 1.0 for pipe failure was then used, with a probability of 0.5 given SCRAM. A probability of 0.1 for early turbine failure was then used.

In the case of the PWR, however, the MFW pumps and main turbine have tripped due to the high water indication. As a result, steam flow would then be directed via the bypass valve to the main condensor. The potential for water hammer still may exist, but the potential for pipe rupture is lowered to 0.5. This is reflected in Figure 2.1.

In this case the steam generator level appears to stabilize at 96.6 percent full. Spill-over of water therefore does not occur. However, given the excessive moisture carry-over, the potential for water condensation and collection followed by water hammer in the steam lines still exists. A probability of pipe break due to water hammer of 0.5 will be used here, down from 1.0, to reflect the fact that no spillover will occur and that the reactor is tripped from only 5 percent power. PNL feels this is highly conservative, but justified given the level of uncertainty at this time. This value should be updated as more realistic modeling of pipe behavior becomes available.

The frequency of inducing a steam line break due to overfill with an initiating frequency of 1.4E-03/py is then reduced to an estimated 7.00E-05/py. This then represents a probability of inducing a steam line break given overfill of (0.1x0.5) = 0.05, with the small assumed value for operator error contributing the most impact on reducing the probability of this scenario progressing to a LOCA.

Accident Progression to Core-Melt

Given steam line failure, the accident can progress as a simple cooldown transient, or it can induce a steam generator tube rupture. Both situations will be considered here. The probability of SGTR given steam line break has been addressed by the NRC (U.S. NRC 1985) as part of its evaluation of Unresolved Safety Issues A-3, A-4, and A-5. This probability is put at the following based on observed experience to date:

p (one or more tube rupture) following an MSLB = 0.034

p (2 to 10 SGTRs) following an MSLB = 0.014

p (more than 10 SGTRs) following an MSLB = 0.003.

The probability of a cooldown transient alone is then (1 - 0.034) = 0.966. This will simply be modeled below for a probability of 1.0. The probability of SGTR will then be modeled with a probability of 0.034.

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Cooldown Transient

The WASH-1400 study (U.S. NRC 1975) used the three-loop Surry Westinghouse plant, which is similar to the H. B. Robinson plant in design and capacity, so this PRA is used for this analysis. WASH-1400 gave consideration to the consequences that would follow from ruptures on either the primary or secondary side of a steam generator. Some 30 possible accident sequences were identified, but these all ended in either a rapid cooldown transient or a LOCA.

The transients induced by steam generator failures did not lead to core-melt in the WASH-1400 study but could release activity from the fuel-clad gap due to fuel damage. However, the end result was that steam generator rupture was not identified as an important factor in the risks due to transient events (U.S. NRC 1975).

To be conservative the excessive cool-down transient will be modeled here with the event for steam line break.

The ORNL Precursor Study (Minarick and Kukielka 1982) indicates that there have been several accidental lifts of steam relief valves. The most similar to this postulated initiator include the incident at Beaver Valley 1, in which the steam dump valves failed to close following load rejection (NSIC 148764), and the incident at Crystal River, in which an excessive cooldown rate resulted from loss of ICS power, turbine trip, and 50 percent opening of the atmospheric dump valves (NSIC 123150).

Both initiators were modeled as a steam line break in the Precursor Study, using the event tree shown in Figure 2.2. This figure ignores initiating event frequency for the moment, showing only the conditional probability of core damage given the initiating steam line break, or in this case the relief valve lift.

The ORNL analysis determined that the prevention of core damage would require steam generator isolation, auxiliary heat removal, high pressure injection (HPI), and long term cooling. The operation of the HPI system with borated water is needed to prevent a return to power as the overcool progresses. The potential for a PORV lift and failure to reseat given continued operation of the HPI is also considered.

Note that the failure to isolate the steam generator is the dominant failure sequence in the ORNL analysis. In this case, no credit is taken for mitigating the consequences of the overcool by injection of the borated water. In addition, ORNL considered severe core damage possible with a stuck rod given steam generator isolation and HPI.

INPO re-analyzed this accident sequence to identify conservative assumptions on the part of ORNL (INPO 1982). The event tree developed by INPO is shown in Figure 2.3. INPO observed that the HPI system is designed to borate the reactor even in the case in which all steam generators are not isolated. Credit was then given for HPI operation even if isolation fails. In addition, with isolation, INPO considered operation of the auxiliary feedwater sufficient to prevent core damage. Thus failure of the HPI function would not be expected to result in core damage if the generator were isolated.

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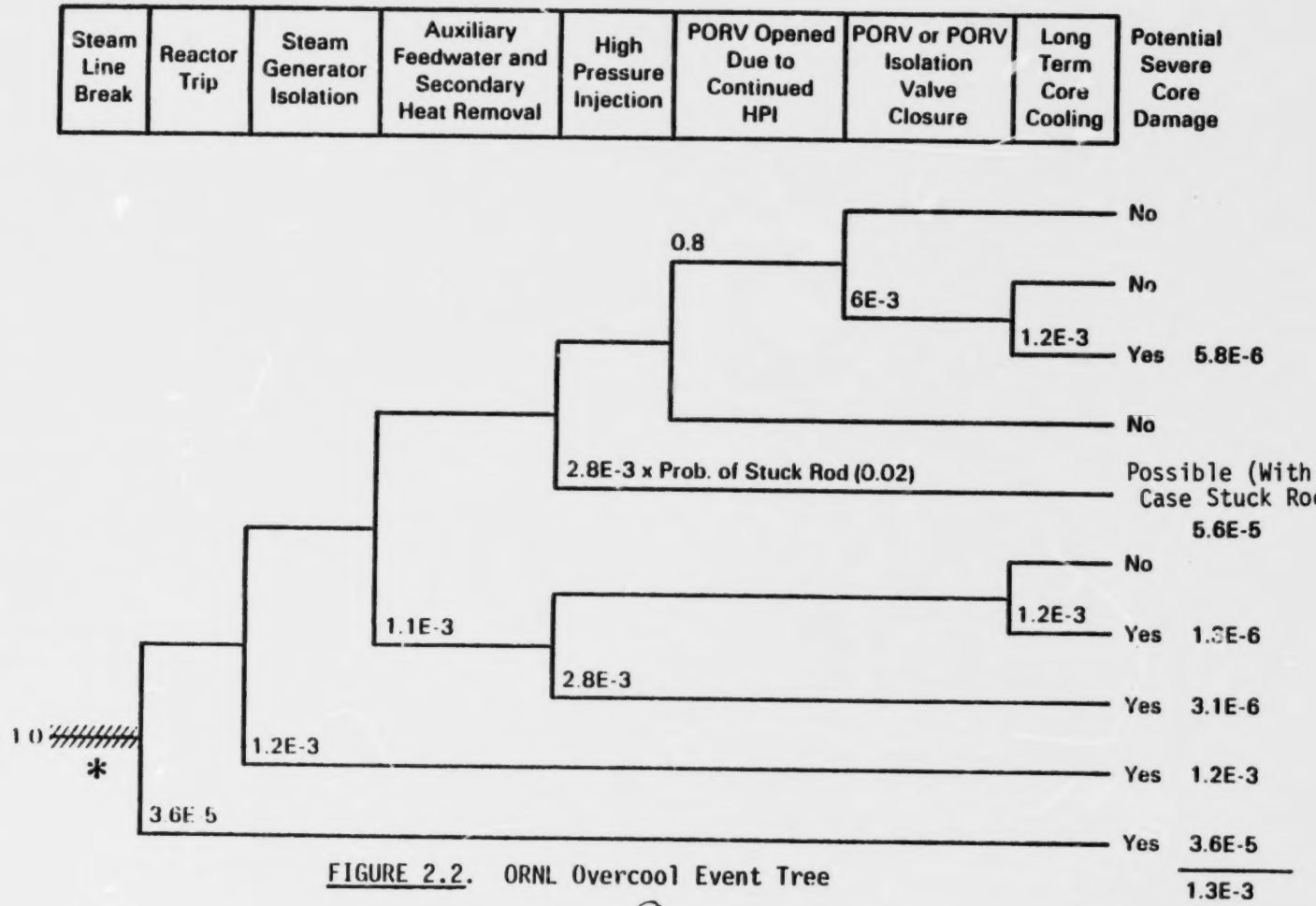

FIGURE 2.2. ORNL Overcool Event Tree

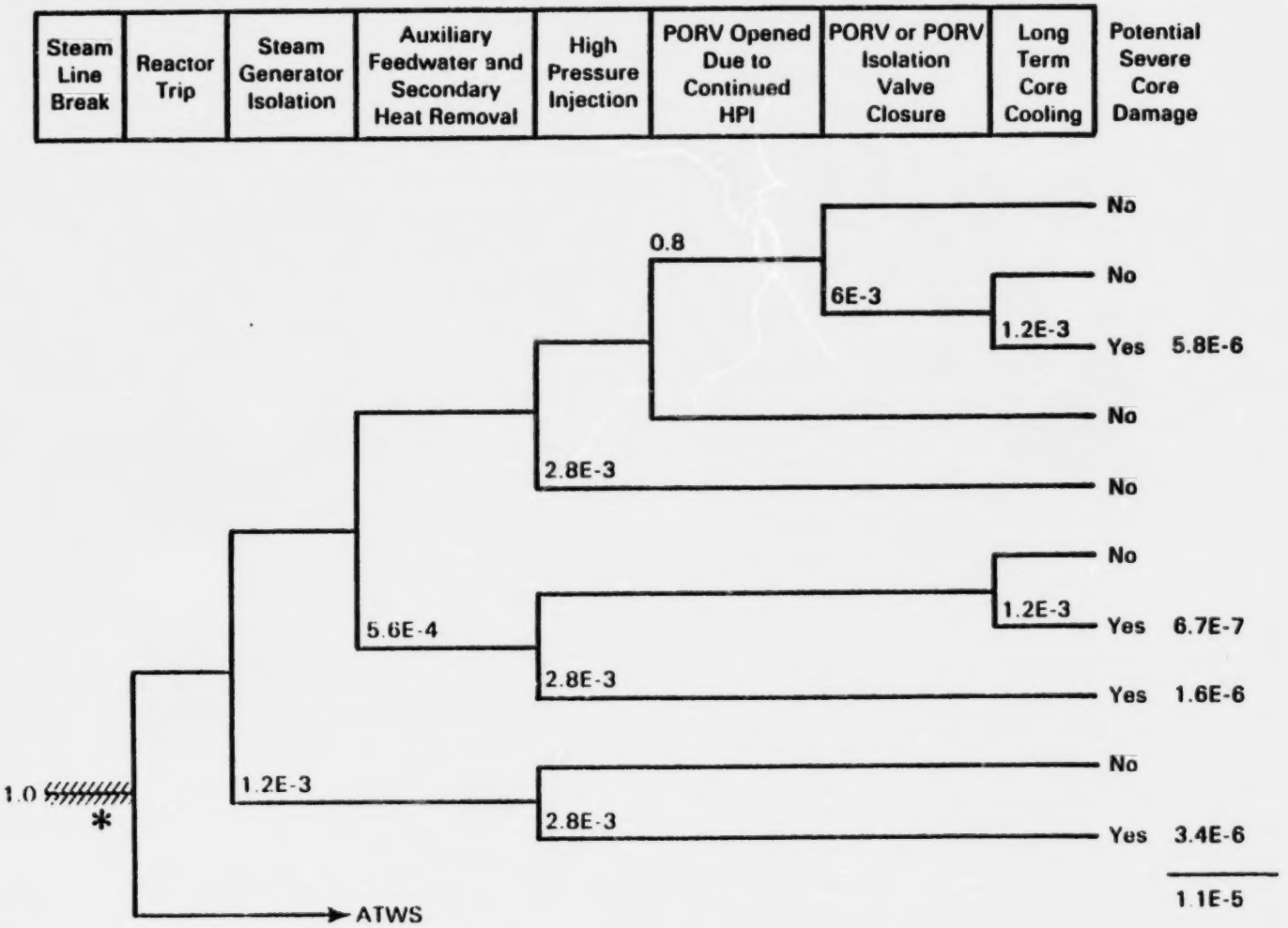

FIGURE 2.3. Modified INPO Overcool Event Tree

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In addition, the possibility of a stuck rod is omitted in the INPO analysis because it is not considered to be significant if complete steam generator isolation has occurred. Also, INPO does not assign core damage directly to the reactor trip failure sequence (ATWS). It is thought that credit should be taken for operation of mitigating systems that can prevent core damage given ATWS. This, however, is not developed here because it is not a dominant contributor to core damage.

Finally, INPO assigned a lower probability (2.1E-04) for failure of the auxiliary feedwater than did the ORNL analysis, noting that the ORNL value of 1.1E-03 was based on system degradation that should not have been counted at system failures. After ACRS review, a value of 5.6E-04 has been proposed.

The net result is an estimated conditional probability of core damage of 1.1E-05 given the valve lift. This compares to the 1.3E-03 value estimated in the ORNL study. The INPO study is thought to correct the uncertainties an deficiencies in the ORNL study and thus will be used here.

For the purposes of this examination, the initiating event will be taken as given.

The specific steps of the tree are discussed further below.

Reactor Trip. The pressure transient and turbine trip caused by the erroneous valve lift will also generate the reactor trip signal. The probability of trip failure is placed at 3.6E-05 in the ORNL report.

If this were assumed to lead to core damage, as it was in the ORNL report, this anticipated transient without SCRAM (ATWS) sequence would be the dominant sequence on the new event tree. However, the INPO report again points out that trip failure will not necessarily lead to core damage. The operation of other systems to mitigate the ATWS is thought by INPO to reduce this sequence to a nondominant contributor to core damage. In this case, HPI of borated water could reduce reactivity increases, and successful steam generator isolation would likely restore feedwater flow. The INPO observation that this sequence would not proceed directly to core damage will therefore be used here, and the ATWS branch will not be fully developed.

Steam Generator Isolation. This step requires both closure of the main steam isolation valve (MSIV) to the affected steam line, and isolation of the feedwater flow. The failure probability of this is set at 1.2E-03 in both the ORNL and INPO studies. The postulated failures causing the inadvertent lift of the relief valve would not impact this probability.

Note that the INPO study points out that the relief valves can also be manually isolated with a block valve. However, it is assumed that the operator would contribute to this term as well as to that above for feedwater isolation, thus not significantly reducing the failure probability.

Auxiliary Feedwater and Secondary Heat Removal. This postulated accident is not thought to affect the performance of the auxiliary feedwater system. Again, the failure probability was originally put at 1.1E-03, and estimated at 2.1E-04 by INPO after an examination of degraded performance. This was later reviewed by the ACRS, with the recommended value of 5.6E-04 used here.

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High Pressure Injection (HPI). In this overcool scenario, injection of borated water will be required primarily for reactivity control in addition to coolant inventory control. The postulated valve lift is not thought to affect the performance of this system. The failure probability of the HPI function is put at 2.8E-03 as per the ORNL and INPO studies.

Power Operated Relief Valve (PORV) Opened Due to Continued HPI. Continued operation of the HPI could cause lifting of the PORV. In accordance with the ORNL and INPO studies, the probability of valve lift is put at 0.8.

PORV or PORV Isolation Valve Closure. Given the lifting of the PORV, the probability of failure to close the valve is 6E-03, as per the previous studies.

Long Term Core Cooling. The failure probability of this function is given in the ORNL study is 1.2E-03.

The predicted probability of core damage, given the initiating MSLB, is then estimated to be 1.1E-05.

The net probability of severe core damage given overfill and steam line break is then 1.1E-05 (obtained from Figure 2.3).

2.3 FREQUENCY OF CORE DAMAGE DUE TO OVERFILL SEQUENCE 1 WITH STEAM LINE BREAK

The predicted frequency of core damage due to steam generator overfill is then estimated to be (1.4E-03/py)(0.05)(1.1E-05) = 7.70E-10 py. If no corrections are applied to distinguish between core damage and core-melt, the predicted frequency of core-melt is then 7.70E-10 py.

2.4 PUBLIC RISK DUE TO OVERFILL SEQUENCE WITH STEAM LINE BREAK

Again, this scenario, with overfill and subsequent main steam line break, is expected to lead to core damage only. However, for the purposes of bounding the potential risk to the public, the results above will be associated with core-melt. Accident sequences for the Oconee PRA involving loss of the power conversion system, relief valve closure failure, loss of long term decay heat removal, or loss of high pressure injection are typically associated with PWR release categories 3, 5, and 7, with the probabilities of 0.5, 0.0073, and 0.5, respectively (EPRI and Duke Power 1984). This will be assumed to be applicable here, where loss of decay heat path leads to core damage. The public risk associated with these release categories is then developed in Table 2.2.

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TABLE 2.2. Public Risk Associated with Steam Generator Overfill Sequence 1

Release Category	Probability	Man-rem/release	Frequency, 1/py	Man-rem/py
PWR-3	0.5	5.4E+06	7.70E-10	2.1E-03
PWR-5	0.0073	1.0E+06	7.70E-10	5.6E-06
PWR-7	0.5	2.3E+03	7.70E-10	8.9E-07
TOTAL				2.1E-03

The total predicted exposure due to Overfill Sequence 1 is therefore on the order of 2.1E-03 man-rem/py.

2.5 CORE-MELT DUE TO OVERFILL INDUCED SGTR

As discussed previously, the conditional probability of SGTR given a MSLB will be put at 0.034 based on the considerations presented in NUREG-0844. This represents a potential for single and multiple tube ruptures. The initial plant response to tube ruptures can be modeled as an SBLOCA. The WASH-1400 Surry analysis made no distinction between the two. However, the long-term system response to a SGTR may differ to that for a LOCA in that water released from the break is not available for collection at sumps within the reactor building. Long term recirculation modes may not also be available as they would for most LOCAs.

To model the plant response to a SGTR, PNL examined the specific event tree developed for this purpose for the Zion Westinghouse PWR, and examined how the specific scenerio postulated by INEL might modify plant response from a simple SGTR event. The Zion event tree is shown in Figure 2.4. The resulting dominant sequences and conditional frequencies are then given in Table 2.3. These results are based primarily on an assumed conditional initiating frequency for SGTR of 1/py with electric power supplies in specific configurations of availability. Lower conditional frequencies for SGTR were assumed for configurations when electric power unavailability made key equipment unavailable.

The results predict a core-melt associated with transient-type behavior, all with early core-melt. From Table 2.3, sequences 9, 3, and 7 contribute to core-melts characterized by early melting with containment sprays operating with a frequency of (6.68E-06/py + 2.30E-06/py) = 8.98E-06/py. It will be assumed here that this value will be associated with WASH-1400 release category 5. Sequences 11, 5, and 8 contribute to early melting with failure of the containment sprays with a frequency of (4.51E-09/py + 2.05E-07/py) = 2.10E-07/py. This value will be associated here with WASH-1400 release category 2.

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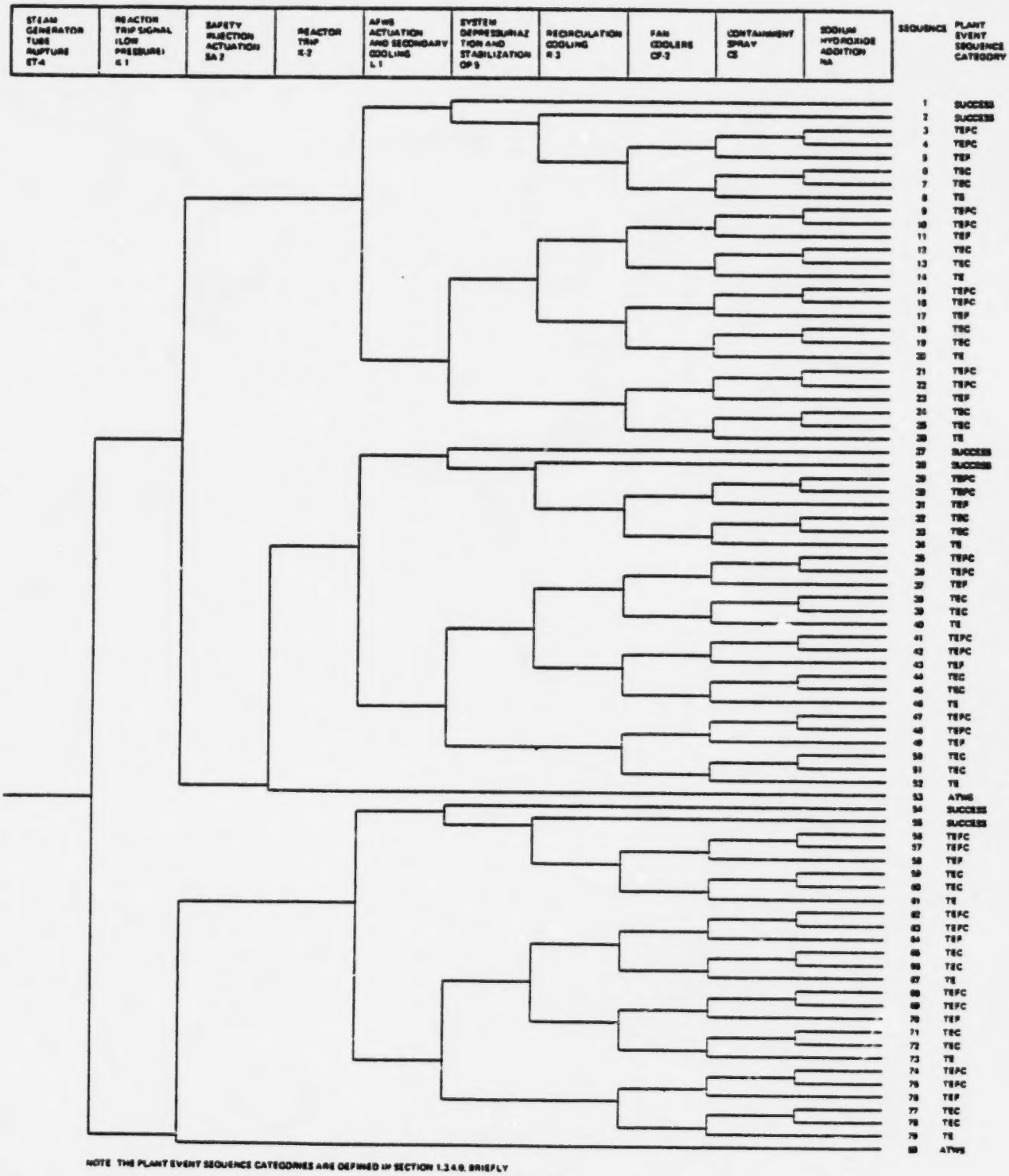

FIGURE 2.4. Steam Generator Tube Rupture Event Tree for Zion Westinghouse PWR

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TABLE 2.3. Frequency of Core-Melt Given SGTR for Zion Westinghouse PWR (U.S. NRC 1981b)

Plant Event Sequence Category[2]	Conditional Frequency	Dominant Sequences			
		Sequence and AC Buses Available		Failed Branch Points	Conditional Frequency
		Bus No. 14	Seq.		
TEFC	6.68-6	7,8,9	9	L-1	4.10-6
		7,8,9	3	OP-5, R-3	2.28-6
TEF	4.51-9	7,8,9	11	L-1, CS	2.26-10
		7,8,9	5	OP-5, R-3, CS	1.25-10
		7,8	5	R-3, CS	3.61-9
TEC	2.30-6	7	7		2.16-7
TE	2.05-7	None	8		1.75-7
		7	8		1.75-7
ATWS	1.02-10	7,8,9	80	K-1, SA-2	5.12-11
		7,8,9	53	SA-2, K-2	5.12-11

NOTES: 1. Dominant sequences are shown with respect to 6-hour electric power bounding model results.

2. The Plant Event Sequence Categories are defined in Section 1.3.4.0 of Zion PRA.

A - Large LOCA behavior E - Early melt F - Fan coolers are operating
S - Small LOCA behavior L - Late melt C - Containment sprays are operating
T - Transient behavior

3. Values are presented in abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

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The total frequency is then (8.98E-06/py + 2.10E-07/py) = 9.19E-06/py given an SGTR initiation frequency of approximately 1/py. The conditional probability of core-melt given SGTR is then taken as 9.19E-06, or approximately 1E-05, given SGTR. The ATWS sequence does not contribute significantly to this and is thus dropped from further consideration.

This compares favorably to the results of NUREG-0844 (U.S. NRC 1985), where the total contribution to core-melt from SGTR was put at 1.5E-07/py. With an estimated tube rupture frequency of 2E-02/py, this effectively gives a conditional probability of core-melt of 7.5E-06 given SGTR, or again approximately 1E-05, given SGTR.

In both the Zion PRA and the NUREG-0844 report, a number of failure sequences involving SGTR as the initiating event are evaluated that are not specifically applicable here. Successful recovery from a simple SGTR event centers on the ability to isolate the affected steam generator and on depressurization of the reactor cooling system (RCS) before the water inventory in the reactor water storage tank (RWST) is exhausted. Many scenarios for single and especially multiple tube rupture events postulate the lifting and sticking open of steam generator relief valves due to the large pressure spike seen by the secondary side on rupture of the tubes. However, in this case the scenario is driven by an assumed steam line break on the secondary side, making the lift of relief valves unlikely. Failure to isolate the SG due to rupture of the steam line inboard MSIV is also calculated in the NUREG-0844 report, but this initiating frequency is quite low, resulting in a small contribution to the total core-melt frequency for MSLBs inboard of the MSIV.

In this case, however, the potential for a steam line break inboard of the MSIVs may be higher. The analysis here assumes that a steam line break occurs with a high probability given that the overfill occurs. If a conservative approach is further taken to assume a 50% probability of MSLB above or below the MSIV, then this scenerio may play a dominant role in the resulting conditional probability of progression to core-melt.

To determine the potential impact of the MSLB location on core-melt frequency, the appropriate scenarios and failure probabilities from NUREG-0844 (Chapter 3.4) were examined, with the results given below. These have also been coupled with the assumed SGTR probabilities.

Case 1: Rupture of Main Steam Line Inboard of the MSIV

Number of SGTRs	Probability of Rupture	Prob of Loss of RWST before RCS Depressurization	Prob of Failure to Isolate SG	Net Core-Melt Probability
1	0.017	1E-03	1	1.7E-05
2 to 10	0.014	1E-02	1	1.4E-04
more than 10	0.003	0.5	1	1.5E-03
Total Probability of Core-Melt Given MSLB Inboard of MSIV				1.66E-03
Conditional Probability of Core-Melt Given MSLB and SGTR				4.87E-02

Case 2: Rupture of Main Steam Line Downstream of the MSIV

Number of SGTRs	Probability of Rupture	Prob of Loss of RWST before RCS Depressurization	Prob of Failure to Isolate SG	Net Core-Melt Probability
1	0.017	1E-04	1E-03	1.7E-09
2 to 10	0.014	1E-03	1E-03	1.4E-08
more than 10	0.003	1E-03	1E-03	3.0E-09
Total Probability of Core-Melt Given MSLB Downstream of MSIV				1.87E-08
Net Probability of Core-Melt Given MSLB and SGTR				5.50E-07

If a 50% probability of MSLB inboard of the MSIVs is used, the conditional probability of core-melt given MSLB and SGTR can then be weighted, giving (0.5)(4.87E-02 + 5.50E-07) = 2.44E-02. This value will be used here.

Comparison to SBLOCA Response

PNL also examined the possibility of modeling the SGTR event with small break LOCA (SBLOCA) event trees from the Surry WASH-1400 PRA. With an initiating frequency of 3E-04/py for S1 (2 to 6 inch) and 1E-03/py for S2 (0.5 to 2 inch) LOCAs, the resulting Surry core-melt frequencies were 6.1E-06/py and 2.3E-05/py, respectively. This yields a conditional probability of core-melt, given LOCA, of 2.03E-02 and 2.3E-02 for S1 and S2 LOCAs, respectively. The net result of the above consideration for SGTR then is to increase the conditional probability of core-melt given SGTR by several orders of magnitude (9.19E-06 to 2.44E-02) by recognizing the potential for the MSLB to occur above the MSIV. The resulting conditional probability is then similar to the probability of progressing to core-melt compared to SBLOCAs.

In this analysis, the frequency of overfill with MSLB and SGTR is then placed at (1.4E-03/py)(0.05)(0.034) = 2.38E-06/py. Using this new initiating frequency for SGTR, the total predicted frequency of core-melt due to this control system failure is then placed at (2.38E-06/py)(2.44E-02) = 5.81E-08 py.

The public dose was estimated by considering that the above core-melt sequences were brought about by failure of the water storage tank inventory, and water not being available from the building sumps. As a result, the containment sprays would also be inoperable. Zion PRA related the SGTR sequences to release categories 2 and 5, but given the above consideration, only release category 2 at 4.8E+06 man-rem/core-melt will be used here. The results are summarized in Table 2.4.

TABLE 2.4. Public Dose Associated with Overfill Sequence 1, MSLB, and SGTR

Release Category	Man/rem Release	A-47 PNL Analysis of Overfill Sequence 1 with SGTR	
		Core-melt/py Best Estimate	Man-rem/py Best Estimate
2	4.8E+06	5.81E-08	2.81E-01

The total public risk is then estimated at 2.81E-01/py.

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2.6 TRANSIENT SHUTDOWN INDUCED BY OVERFILL SEQUENCE 1

Finally, the very act of plant shutdown can present a safety challenge to feedwater and decay heat removal systems. In the BWR, transient shutdowns can represent the primary source for initiating a core-melt sequence. In PWRs, the transients have generally played a less important role in overall risk (Joksimovich 1984), but will be examined here for completeness.

A review of the PRA for the three loop Surry (WASH-1400) and four loop Sequoyah (NUREG/CR-1659; U.S. NRC 1981a) Westinghouse PWRs indicates both use the same basic event tree to measure risk due to transients that may effect the power conversion system (PCS). The PCS represents the pathway for decay heat rejection after plant shutdown. The variables of interest are then:

T2 = transient frequency with loss of main feedwater = 3/py (Sequoyah)
T3 = transient frequency with main feedwater available = 4/py (Sequoyah)
M = probability of PCS failure = 1.0E-02
L = probability of auxiliary feedwater failure = 4E-05.

A T2 transient with failure of the auxiliary feedwater (L) was assumed to lead to core-melt for that particular plant. Note that other plants also require failure of a high pressure injection function before core-melt, given loss of the PCS.

In this case, the initiating frequency of the overfill event is estimated at 1.4E-03/yr. Further, the probability of operator failure to terminate the event was assumed earlier to be 0.1. The frequency of spillover is then placed at 1.4E-04/yr. Having progressed to the point of spillover, however, the potential for loss of the PCS is much higher than reflected above. The probability of 0.5 used earlier for damage to the steam lines will be used here for loss of the PCS given spillover. Since this scenario actually progresses to this point with auxiliary feedwater, no modification to the auxiliary feedwater failure probability is thought to be necessary. The net result is then a core-melt frequency estimate of (1.4E-03/py)(0.1)(0.5)(4E-05) = 2.80E-09/py.

Risk

The Sequoyah PRA concluded that the TML sequence was the only sequence of risk importance, associated entirely with PWR release category 3. The Surry plant associated this primarily with PWR-7, and only 1% with PWR-3 due to different containment designs. The PWR-3 release category is more severe at 5.4E+06 man-rem/core-melt, and thus will be assumed here. The risk due to transient shutdown with this overfill scenario is then assumed to be (2.80E-09/yr)(5.4E+06 man-rem) = 1.5E-02 man-rem/py.

2.7 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR OVERFILL SEQUENCE 1

The total predicted core-melt frequency due to this issue is then the sum of the steam line break, SGTR and transient scenarios, or (7.70E-08/py + 5.81E-08/py) = 6.17E-08/py. The predicted core-melt is then primarily associated with the SGTR assumption.

The total predicted dose associated with this sequence is also the sum of the steam line break and SGTR scenarios, or (2.1E-03 man-rem/py + 2.8E-01 + 1.5E-02 man-rem/py) = 3.0E-01 man-rem/py.

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3.0 STEAM GENERATOR OVERFILL SEQUENCE 2

As with the steam generator overfill Sequence 1, this sequence also involves failures that result in increased feedwater flow rates. In this case, failures in the feedwater level indicators or controller are postulated. These will be developed more fully below.

3.1 INITIAL PLANT CONDITIONS FOR STEAM GENERATOR OVERFILL SEQUENCE 2

The initial plant conditions assumed by INEL for this sequence are the plant at 67 percent reactor power with the rod control in manual and all other control systems in automatic.

A failure in the feedwater level or control system causing an increased feedwater flow and loss of the high level feedwater trip was then assumed. In this scenario, the steaming rate from the affected steam generator is projected to eventually come into equilibrium with the higher feedwater flow rate, with reactor power increasing to 1740 MW. The water level in the affected steam generator also appears to stabilize; however, steam quality degrades to 13 percent, which indicates that severe moisture carry-over would occur.

The sequence of events postulated by INEL and the required time for each event to take place due to the feedwater failure are shown in Table 3.1.

TABLE 3.1. Sequence of Events for Steam Generator Overfill Sequence 2

Time (s)	Event
0.0	Transient initiated by control or level failure with loss of high trip
7.3	MFW valve wide open
146.0	Oscillations in steam generator mass flow rates stabilize in a smooth, asymptotically increasing rate
167.0	Minimum pressurizer pressure reached
250.0	Primary average temperatures in new equilibrium
300.0	Liquid carry-over peaked at 19.2 percent
400.0	Analysis terminated in steady state with 19.2 percent liquid carry-over fraction, steam quality at 13 percent

The accident initiator postulated by INEL again involves no initial failure of the primary containment or cooling of the core.

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3.2 ACCIDENT PROGRESSION ANALYSIS FOR STEAM GENERATOR OVERFILL SEQUENCE 2

The analysis of this steam generator overfill will be similar to the previous examination in that the potential for inducing a steam line break and SGTR will be of interest. However, in the first scenario the main feedwater flow tripped on the high level, with auxiliary feedwater continuing the overfill. In this case, the main feedwater turbines remain on and could possibly be impacted by the degrading steam quality. The event tree as developed to consider LOCA in the BWR will then be applicable. Figure 3.1 presents this event tree. The various steps are developed below.

Initiating Event. The initiating event for this scenario is the feedwater control opening the main feedwater valve, followed by feedwater trip and continued overfill with the auxiliary feedwater. The median value given for this by INEL is 5.4E-08/py, with an upper bound of 5.5E-07/py. For the purposes of this examination, the INEL estimate of 5.4E-08/py will be used.

Operator Isolates Feedwater. The transient can be terminated by the operator simply isolating the feedwater flow. In this case, however, the reactor is at 67 percent power and in fully automatic feedwater control. In addition, the failures postulated for the feedwater system present conflicting readings to the operator, as was the case for level indicator failure in the BWR. The potential for correct interpretation and action on the part of the operator is then expected to be reduced somewhat compared to the previous overfill scenario.

The value of 0.517 for operator failure was estimated for the BWR. Note, however, that in the PWR spillover does not occur, although steam quality again deteriorates rapidly. A value of 0.5 is proposed here for the PWR given the similar failure mechanisms to the BWR case, but with spillover never actually occurring. This value is thus thought to be more conservative than the 0.517 used in the BWR examination.

Feedwater Turbine Failure. As in the BWR analysis, the potential exists for the excessive moisture carry-over to damage the steam-driven feedwater turbine and end the overfill. This again is put at 0.1. The probability of the overfill continuing is then put at 0.9.

Main Turbine Failure and SCRAM. As with the previous analyses, a probability of 0.1 of main turbine failure and shutdown will be assumed, with a probability of 0.9 that the overfill would continue with the reactor at power.

Note that as with the previous PWR overfill scenario, spillover of water into the steam lines does not occur with the reactor at power. A reactor SCRAM, however, could produce a spillover given continued feedwater flow after reactor shutdown and reduced steaming, although feedwater flow will also decay after shutdown.

Steam Line Break. As in the previous analysis, the potential for steam line break given reactor shutdown and spillover will be estimated at 0.5 for water hammer plus 0.001 for static load. Given continued operation, the probability for water hammer-induced pipe break will be put at 0.5, reduced from the 1.0 used for the BWR where spillover of water actually occurred at power, since in this case, spillover will not occur if operation continues.

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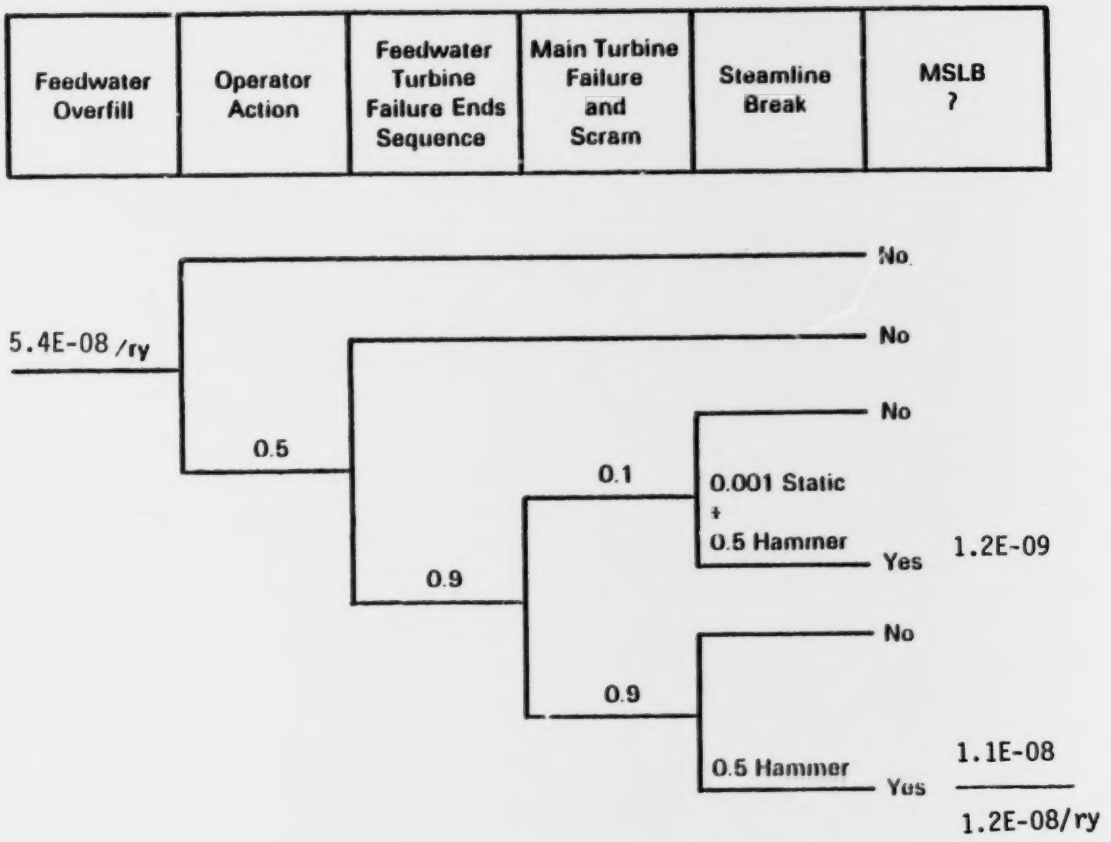

FIGURE 3.1. Feedwater Overfill Sequence 2: Progression to Steam Line Break

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The frequency of inducing a steam line break due to overfill with an initiating frequency of 5.4E-08/py is then reduced to an estimated 1.24E-08/py. This then represents a probability of inducing a steam line break of 0.23 given the initiating event, i.e., (0.23)(5.4E-08/py) = 1.24E-08/py.

Steam Line Break Event Tree. The reactor response to this steam line break will be identical to that developed for the previous steam generator overfill Sequence, 1 as shown in Figure 2.2. This figure estimated the probability of core damage given the steam line break of 1.1E-05.

3.3 FREQUENCY OF CORE DAMAGE DUE TO OVERFILL SEQUENCE 2 WITH STEAM LINE BREAK

The predicted frequency of core damage due to steam generator overfill is then estimated to be (5.4E-08/py)(0.23)(1.1E-05) = 1.37E-13/py. If no corrections are applied to distinguish between core damage and core-melt, the predicted frequency of core-melt is then 1.37E-13/py. This value could be considered negligible, but will be carried through to public risk for completeness.

3.4 PUBLIC RISK DUE TO OVERFILL SEQUENCE 2 WITH STEAM LINE BREAK

As with the overfill Sequence 1, accident sequences for the Oconee PRA involving loss of the power conversion system, relief valve closure failure, and loss of long-term decay heat removal or loss of high pressure injection are typically associated with PWR release categories 3, 5, and 7 with the probabilities of 0.5, 0.0073, and 0.5, respectively. This will be assumed to be applicable here even though it is expected to give a highly conservative measure of risk associated with a core damage event. The public risk associated with these release categories is developed in Table 3.2.

TABLE 3.2. Public Risk Associated with Steam Generator Overfill Sequence 2

Release Category	Probability	Man-rem/release	Frequency, 1/py	Man-rem/py
PWR-3	5.0E-01	5.4E+06	1.37E-13	3.7E-07
PWR-5	7.3E-03	1.0E+06	1.37E-13	1.0E-09
PWR-7	5.0E-01	2.3E+03	1.37E-13	1.6E-10
TOTAL				3.7E-07

The total predicted exposure due to Overfill Sequence 2 with MSLB is then 3.7E-07 man-rem/py. Again, this value is considered negligible, but will be used for completeness.

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3.5 CORE-MELT DUE TO OVERFILL INDUCED SGTR FOR OVERFILL SEQUENCE 2

As with Sequence 1, the conditional probability of core-melt given SGTR is again taken as 2.44E-02.

In this sequence, the frequency of overfill with MSLB and SGTR is put at (5.4E-08/py)(0.23)(0.034) = 4.22E-10/py. Using this new initiating frequency for SGTR, the total predicted frequency of core-melt due to this control system failure is then put at (4.22E-10/py)(2.44E-02) = 1.03E-11/py.

The public dose is again estimated by release category 2 at 4.8E+06 man-rem/core-melt. The results are summarized in Table 3.3.

TABLE 3.3. Public Dose Associated with Overfill Sequence 2, MSLB, and SGTR

		A-47 PNL Analysis of Overfill Sequence 2 with SGTR	
Release Category	Man-rem Per Event	Core-melt/py Best Estimate	Man-rem/py Best Estimate
2	4.8E+06	1.03E-11	4.9E-05

The total public risk associated with Overfill Sequence 2 progressing to SGTR is then estimated at 2.02E+0/py.

3.6 TRANSIENT SHUTDOWN INDUCED BY OVERFILL SEQUENCE 2

As with Overfill Sequence 1, the very act of plant shutdown can present a safety challenge to feedwater and decay heat removal systems. The same TML core-melt sequence is thought to be of primary interest here, with the necessary modifications for Overfill Sequence 2.

In this case, the initiating frequency of the overfill event is put at 5.4E-08/yr. The conflicting signals to the operator were thought to increase the potential for operator failure to terminate the sequence at a probability of 0.5.

As with Overfill Sequence 1, having progressed to the point of spillover, the potential for loss of the PCS is set at 0.5 to reflect the potential for damage to the steam lines and condensor. This compares to the 1E-02 probability used in the Surry PRA. Again, no modification to the auxiliary feedwater failure probability is thought to be necessary at 4E-05. The net result is then a core-melt frequency estimate of (5.4E-08/yr)(0.5)(0.5)(4E-05) = 5.40E-13/yr.

Risk

The PWR-3 release category at 5.4E+06 man-rem/core-melt will again be assumed here. The risk due to transient shutdown with this overfill scenario is then set at (5.40E-13/yr)(5.4E+06 man-rem) = 2.9E-06 man-rem/py.

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3.7 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR OVERFILL SEQUENCE 2

The total predicted core-melt frequency due to this issue is then the sum of the steam line break and SGTR scenarios, or 1.37E-13/py + 1.03E-11/py + 5.40E-13/py = 1.10E-11/py. The predicted core-melt is again primarily associated with the SGTR assumption.

The total predicted dose associated with this sequence is then also the sum of the steam line break, SGTR, and transient scenarios: 3.7E-07 man-rem/py + 4.9E-05 man-rem/py + 2.9E-06 man-rem/py = 5.2E-05 man-rem/py.

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4.0 REACTOR COOLANT SYSTEM OVERCOOL SEQUENCE 1

This sequence involves a failure of a steam dump valve opening, initiating an overcooling of the primary coolant system. The initial conditions and plant response are discussed below.

4.1 INITIAL PLANT CONDITIONS FOR OVERCOOL SEQUENCE 1

The initial plant conditions assumed by INEL for this sequence are the plant at 102 percent reactor power and all control systems in automatic. The sequence is initiated by failing the steam dump valves to their open position. This failure creates an increase in steam flow, which reduces secondary pressure and causes a pressure surge back into the feed system that trips the main feedwater pump on low feedline pressure. This leads to a turbine trip, a reactor trip, and closure of the turbine stop valve.

The high initial power level was assumed in order to place the plant closer to the setpoints that actuate a reactor trip. The INEL analysis indicated that a plant overcool would not occur if the reactor did not trip. The sequence of events postulated by INEL and the required time for each event to take place due to the steam dump valve failing open are shown in Table 4.1.

TABLE 4.1. Sequence of Events for Reactor Coolant System Overcool Sequence 1

Time (s)	Event
0.0	Steam dump valves opened
2.1	Main feedwater pump tripped on low feedline pressure Turbine tripped, reactor tripped, and turbine stop valve closed
5.0	Low RCS pressure
45.0	Low pressurizer level, pressurizer heaters tripped off
90.0	Pressurizer emptied
95.0	Void started to form in upper plenum
110.0	Void started to form in reactor upper head region, void from upper plenum began to transfer to the upper head region
240.0	End of calculations

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The accident initiator as postulated by INEL thus involves no initial failure of the primary containment or cooling of the core. In fact, the overcooling of the primary system has led to significant shrinkage of the primary coolant, with subsequent draining of the pressurizer.

4.2 ACCIDENT PROGRESSION ANALYSIS FOR OVERCOOL SEQUENCE 1

Again, the WASH-1400 study concluded that ruptures of the steam generator shell causing transient overcools were not sufficient in themselves to lead to core-melt accidents. As with the previous scenarios examined, however, the lift of the steam relief valve can be modeled as a steam line break, which was considered a precursor to core damage in the ORNL Precursor Study. Again, to be conservative, the valve lift will first be modeled as a main steam line break (MSLB). The probability of inducing a SGTR given the MSLB will then also be considered.

The other consideration is for the overcool scenario to lead to pressurized thermal shock and main vessel rupture that would preclude recovery from the accident. To investigate this, the potential for pressurized thermal shock and vessel failure was also examined to the extent possible at this time. The PTS program at ORNL is only now in the process of issuing a draft analysis of PTS events in the H. B. Robinson plant, so it is uncertain which of the scenarios examined best predicts the potential for vessel failure as it applies to the A-47 program. The preliminary information available indicates that the cooldowns presented here would be bounded by the scenario involving lift of all five steam line valves and pressurization of the vessel with high pressure injection. The ramifications of PTS will be discussed after MSLB and SGTR are examined below.

Initiating Event. The failure mode that initiates this scenario is inadvertent opening of the steam dump valve while the reactor is at power. The median frequency of this failure is calculated by INEL to be 2.6E-07/py. This is based on an examination of the possible failures in the temperature instrumentation and solid-state control elements for the steam dump valve controller. The 90th percentile upper bound was set at 1.4E-06/py.

For the purposes of this examination, the INEL estimate of 2.6E-07/py will be used as representative of overcool from failures in the valve controller.

This issue is assumed to apply to the PORVs or steam dump valves downstream of the MSIVs rather than the safety relief valves on the secondary side. These are the only valves being instrumented for lift and closure, the SRVs being seated by spring pressure only. The PORVs also have operator actuated block valves upstream of the PORVs, making isolation of a failed PORV possible.

Continuation of this scenario as a MSLB then requires PORV lift, failure of the PORV to close, and failure of the operator to actuate the block valve or failure of the block valve itself to close on demand. An examination of Safety Issue 70, PORV and Block Valve Reliability, and Task Action Item II.K.3.2 indicates that the accepted values for these variables are as follows:

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Frequency of PORV lift = 1/py

p(failure of the PORV to close) = 0.02/demand

p(failure of the operator to actuate the block valve) = 0.05

p(failure of one block valve) = 0.005/demand.

In this case, it will be assumed that the failure postulated by INEL already precludes successful electronic signals for closure of the PORV. Thus the alternate failure pathway is via the operator or block valve, giving a failure probability of (0.05 + 0.005) = 0.055. Note that the 0.005 failure for one block valve should be multiplied by the total number of block valves on the generator, assumed to be 1 here.

Progression of the scenario as an MSLB then has the frequency of (2.6E-07/py)(0.055) = 1.43E-08/py.

4.3 FREQUENCY OF CORE-MELT DUE TO OVERCOOL SEQUENCE 1 AND MSLB

Referring back to Figure 2.3, the probability of core damage given MSLB was put at 1.1E-05. The predicted frequency of core damage due to advertent valve lift is then estimated to be (2.6E-07/py)(0.055)(1.1E-05) = 1.57E-13/py. If no corrections are applied to distinguish between core damage and core-melt, the predicted frequency of core-melt is then 1.57E-13/py. Values this low are again considered to be negligible, but will be included for completeness.

4.4 PUBLIC RISK DUE TO OVERCOOL SEQUENCE 1 AND MSLB

Accident sequences for the Oconee PRA involving loss of the power conversion system, relief valve closure failure, and loss of long term decay heat removal or loss of high pressure injection are typically associated with PWR release categories 3, 5, and 7, with the probabilities of 0.5, 0.0073, and 0.5, respectively. This will be assumed to be applicable here. The public risk associated with these release categories is developed in Table 4.2.

TABLE 4.2. Public Risk Associated with Reactor Coolant System Overcool Sequence 1

Release Category	Probability	Man-rem/release	Frequency, 1/py	Man-rem/py
PWR-3	0.5	5.4E+06	1.57E-13	4.2E-07
PWR-5	0.0073	1.0E+06	1.57E-13	1.1E-09
PWR-7	0.5	2.3E+03	1.57E-13	1.8E-10
TOTAL				4.2E-07

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The total predicted exposure due to Overfill Sequence 1 is therefore 3.9E-08 man-rem/py.

4.5 CORE-MELT DUE TO OVERFILL INDUCED SGTR FOR OVERCOOL SEQUENCE 1

As with Sequence 2, the conditional probability of core-melt given SGTR is again 2.44E-02.

In this sequence, the frequency of overfill with MSLB and SGTR is set at (2.6E-07/py)(0.055)(0.034) = 4.86E-10/py. Note that the relief valve postulated to lift in this scenario is again inboard of the MSIV. The potential then exists for failures to isolate the steam generator, as in the previous overfill scenarios where a MSLB was postulated to occur above the MSIV. As a result, the same probability of progression to core-melt will be used here, given SGTR. Using this new initiating frequency for SGTR, the total predicted frequency of core-melt due to this control system failure is then put at (4.86E-10/py)(2.44E-02) = 1.19E-11/py.

The public dose is again estimated by assuming PWR release category 2 at 4.8E+06 man-rem/core-melt. The results are summarized in Table 4.3.

TABLE 4.3. Public Dose Associated with Overcool Sequence 1, MSLB, and SGTR

		A-47 PNL Analysis of Overfill Sequence 2 with SGTR	
Release Category	Man-rem per event	Core-melt/py Best Estimate	Man-rem/py Best Estimate
2	4.8E+06	1.19E-11	5.7E-05

The total public risk associated with Overcool Sequence 1 progressing to SGTR is then estimated at 5.7E-05/py.

4.6 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR OVERCOOL SEQUENCE 1

The total predicted core-melt frequency due to this issue is then the sum of the steam line break and SGTR scenarios, or (1.57E-13/py + 1.19E-11/py) = 1.21E-11/py. The predicted core-melt is again primarily associated with the SGTR assumption.

The total predicted dose associated with this scenario is then also the sum of the steam line break and SGTR scenarios: 4.2E-071 man-rem/py + 5.7E-05 man-rem/py = 5.7E-05 man-rem/py. Again, these values are considered negligible, but are given for completeness.

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The other consideration is for the overcool scenario to lead to pressurized thermal shock and main vessel rupture that would preclude recovery from the accident. To investigate this consideration, the potential for pressurized thermal shock and vessel failure was also examined to the extent possible. The PTS program at ORNL is only now in the process of issuing a draft analysis of PTS events in the H. B. Robinson plant, so it is uncertain which of the scenarios examined best predicts the potential for vessel failure as it applies to the A-47 program. The preliminary information available indicates that the cooldowns presented here would be bounded by the scenario involving lift of all five steam line valves and pressurization of the vessel with high pressure injection.

The overall frequency of overcool initation and vessel rupture for this scenario would then be estimated as the initiation frequency (2.6E-07/py) times some reasonable probability that the operator would fail to halt the PTS event by isolating the affected steam generator or reduce HPI flow (0.1), times the probability of PTS induced vessel rupture. The overcool progresses to 200°F and pressurization to 1650 psi after 2 hours, giving an estimated probability of vessel failure of 1E-04 based on the ORNL work. This gives an estimate of core-melt frequency of 2.6E-12/py, or approximately 24 percent of the core-melt frequency attributed to progression to MSLB and SGTR. In any event, the numbers are insignificant.

The indications are, however, that vessel failure probabilities are highly sensitive to the assumed vessel weld composition and neutron fluence. These calculations were done with a conservative reference nil ductility transition temperature of 270°F. Using actual H. B. Robinson plant data, this transition temperature is actually 130°F, and the failure probability for this scenario is reduced to less than 1E-06. Using actual data, the contribution to core-melt frequency from PTS then becomes even more insignificant compared to MSLB and progression to SGTR.

The latter estimate is thought to be the best measure of risk due to PTS induced failures in the A-47 program. It is appropriate for conservative calculations to be used within the PTS program for evaluation and resolution of the PTS issue. However, these conservatisms should not be transferred into related issues if undue weighting or influence of specific safety concerns is to be avoided in a calculation of relative risk. This best estimate then indicates that PTS plays a minimal role in core-melt and risk in the A-47 analysis of this overcool scenario in the Westinghouse H. B. Robinson PWR.

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5.0 REACTOR COOLANT SYSTEM OVERCOOL SEQUENCE 2

This sequence involves a failure of a steam line power operated relief valve (PORV) opening, initiating an overcooling of the primary coolant system.

5.1 INITIAL PLANT CONDITIONS FOR OVERCOOL SEQUENCE 2

The initial plant conditions assumed by INEL for this sequence are the reactor in a hot shutdown condition with the reactor coolant pumps operating and an RCS temperature of 547°F. Rod control, pressure control, and feedwater control are in manual with all other parameters being controlled automatically. The transient was initiated by failing the steamline PORVs open.

The sequence of events as postulated by INEL and the required time for each event to take place due to steam generator tube rupture are shown in Table 5.1.

TABLE 5.1. Sequence of Events for Reactor Coolant System Overcool Sequence 2

Time (s)	Event
0.0	Steamline PORVs begin to open
1.0	Steamline PORVs are fully open
3.0	Combined PORV flows peak at 832 lbm/s
13.0	Primary to secondary heat transfer rate increased from 8 MW to 260 MW
14.9	Safety injection actuation signal (SIAS) on one out of three high steam headers to steamline pressure differential (100 psid)
	Main feedwater (MFW) pumps tripped. Motor driven auxiliary feedwater (MAFW) pumps initiated
15.0	Steamline PORV combined flow at 655 lbm/s
113.0	Primary to secondary heat transfer rate began decreasing due to change from subcooled to saturated nucleate boiling
230.0	Reactor vessel downcomer temperature had decreased 100 F from the initial value
250.0	End of calculation

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The accident initiator postulated by INEL thus involves no initial failure of the primary containment or cooling of the core. In fact, the overcooling of the primary system has led to significant shrinkage of the primary coolant, with subsequent draining of the pressurizer.

The results of the analysis performed by INEL demonstrate that when the reactor is subcritical, a failure that results in the removal of energy in excess of that being added by the RCPs will cause a RCS cooldown.

Following is a detailed discussion of the analysis done on accident progression.

5.2 ACCIDENT PROGRESSION ANALYSIS FOR REACTOR COOLANT SYSTEM OVERCOOL SEQUENCE 2

As with the previous overcool scenario examined, the lift of the steam relief valve can be modeled as a steam line break, which was considered as a precursor to core damage in the ORNL Precursor Study. Again, to be conservative, the valve lift will first be modeled as a main steam line break (MSLB). The probability of inducing an SGTR given the MSLB will then also be considered.

The potential for main vessel rupture with the overcool event was also considered, but like Overcool Sequence 1, the probability of this was put at zero.

This scenario is very similar to reactor coolant system Overcool Sequence 1. In both cases, failing open of steam relief valves caused excessive reactor coolant system overcool. The basic difference is that in Overcool Sequence 1 the plant was at 102 percent power, while in Sequence 2 it was in hot shutdown. Both of these scenarios have been modeled as a steam line break in the ORNL Precursor Study (Minarick and Kukielka 1982) and also in the INPO review of ORNL's study (INPO 1982).

The generic event tree shown in Figure 5.1 used for this sequence will be similar to that used for Overcool Sequence 1, as shown in Figure 4.2.

The basic sequence of events following the initiation of this transient is similar to Overcool Sequence 1. The only difference is that no reactor trip is required (the plant is already shut down). The net result, however, is the same, giving a probability of core damage of 1.1E-05 given the initiating event.

Initiating Event. The failure mode that initiates this scenario is the failing open of the steamline relief valves. The median frequency of this failure is calculated by INEL to be 1.8E-02/py. This estimate is based on the examination of the possible failures in the solid state devices used in the steam dump controller and in the turbine EHC or in the steamline PORV control circuit. For this study the INEL estimate of 1.8E-02/py will be used. The 90th percentile upper bound is set by INEL at 5.0E-02/py.

This scenario again assumes a control system failure opening a PORV on the secondary side. The same failure probability of (0.05 + 0.005) = 0.055 for the operator or block valve to fail to isolate the PORV is then used.

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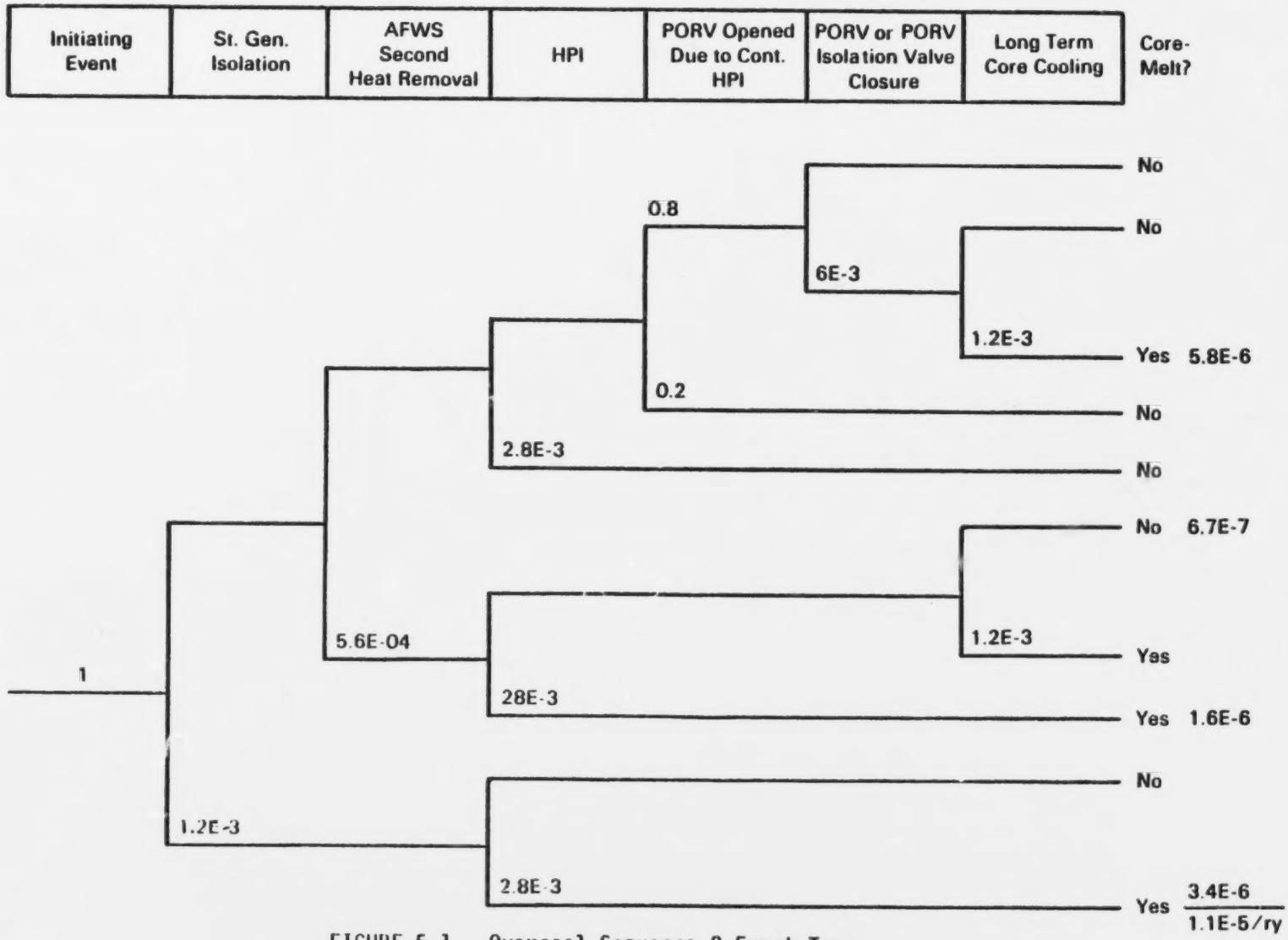

FIGURE 5.1. Overcool Sequence 2 Event Tree

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Note again that the 0.005 failure for one block valve should be multiplied by the total number of block valves on the generator, assumed to be 1.

Progression of the scenario as an MSLB then has the frequency of (1.8E-0E-02/py)(0.055) = 9.90E-04/py.

5.3 FREQUENCY OF CORE-MELT DUE TO OVERCOOL SEQUENCE 2 AND MSLB

The predicted frequency of core damage due to the failing open of the steamline relief valve is estimated to be (1.8E-02/py)(0.055)(1.09E-08) = 1.09E-08/py. If no corrections are applied to distinguish between core damage and core-melt, the predicted frequency of core-melt is then 1.09E-08/py.

5.4 PUBLIC RISK DUE TO OVERCOOL SEQUENCE 2 AND MSLB

Accident sequences for the Oconee PRA involving loss of the power conversion system, relief valve closure failure, and loss of long term decay heat removal or loss of high pressure injection are typically associated with PWR release categories 3, 5, and 7 with the probabilities of 0.5, 0.0073, and 0.5, respectively. This will be assumed to be applicable here. The public risk associated with these release categories is developed in Table 5.2.

TABLE 5.2. Public Risk Associated with Reactor Coolant System Overcool Sequence 2

Release Category	Probability	Man-rem/release	Frequency, 1/py	Man-rem/py
PWR-3	0.5	5.4E+06	1.09E-08	2.9E-022
PWR-5	0.0073	1.0E+06	1.09E-08	8.0E-05
PWR-7	0.5	2.3E+03	1.09E-08	1.3E-05
TOTAL				2.9E-02

The total predicted exposure due to Overfill Sequence 2 is therefore 2.9E-02 man-rem/py.

5.5 CORE-MELT DUE TO OVERFILL INDUCED SGTR FOR OVERCOOL SEQUENCE 2

As with Overfill Sequence 1, the conditional probability of core-melt given SGTR is again 2.44E-02.

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In this sequence, the frequency of overfill with MSLB and SGTR is put at (1.8E-02/py)(0.055)(0.034) = 3.37E-05/py. Using this new initiating frequency for SGTR, the total predicted frequency of core-melt due to this control system failure is then put at (3.37E-05/py)(2.44E-02) = 8.22E-07/py.

The public dose is again estimated by release category 2 at 4.8E+06 man-rem/core-melt. The results are summarized in Table 5.3.

TABLE 5.3. Public Dose Associated with Overcool Sequence 2, MSLB, and SGTR

		A-47 PNL Analysis of Overfill Sequence 2 with SGTR	
Release Category	Man-rem Release per Event	Core-melt/py Best Estimate	Man-rem/py Best Estimate
2	4.8E+06	8.22E-07	3.9E

The total public risk associated with Overcool Sequence 2 progressing to SGTR is then estimated at 3.9E/py.

5.6 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR OVERCOOL SEQUENCE 2

The total predicted core-melt frequency due to this issue is the sum of the steam line break and SGTR scenarios, or (1.09E-08/py + 8.22E-07/py) = 8.33E-07/py. The predicted core-melt is again primarily associated with the SGTR assumption.

The total predicted dose associated with this issue is also the sum of the steam line break and SGTR scenarios: 2.9E-02 man-rem/py + 3.9E man-rem/py = 3.9 man-rem/py.

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6.0 REACTOR COOLANT SYSTEM OVERPRESSURE SEQUENCE 1

This sequence involves failures of the letdown flow and PORV valves in a PWR during cold shutdown, resulting in continued charging and pressurization of the pressure vessel.

6.1 INITIAL PLANT CONDITIONS FOR OVERPRESSURE SEQUENCE 1

The initial plant conditions assumed by INEL were a reactor shutdown with the RCS liquid solid at 100°F and 365 psig. One charging pump was in operation, providing 77 gpm flow throughout the transient. The transient was initiated with a power failure that simultaneously closed the letdown valve and failed one PORV. An additional failure was that the second PORV also failed to open when the pressure exceeded the low temperature mode setpoint of 415 psia.

The sequence of events postulated by INEL and the required time for each event to take place are shown in Table 6.1.

TABLE 6.1. Sequence of Events for Overpressure Sequence 1

Time (s)	Event
0.0	Transient initiated by isolation of letdown valve and failure of one PORV
10.5	Failure of second PORV
105	Calculation terminated

The accident initiator postulated by INEL thus involves no initial failure of the primary containment or cooling of the core.

6.2 ACCIDENT PROGRESSION ANALYSIS FOR OVERPRESSURE SEQUENCE 1

The Westinghouse system uses both centrifugal and reciprocating charging pumps. The two centrifugal pumps provide 2 x 75 = 150 gpm at 2800 psig, and the 1 reciprocating pump provides 98 gpm at 3200 psig. In addition, the following reactor coolant system designs pressures apply:

Hydrostatic test pressure	3110 psig
Design pressure	2485 psig
Safety valves	2485 psig
PORVs	2335 psig

In this scenario defined by INEL, one charging pump is running, supplying 77 gpm. This implies that one centrifugal pump is supplying the feed, with a

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maximum pressure head of 2800 psig. Failure of the letdown and two PORVs would then bring the system past the PORV lift pressure of 2335 psig to the safety valve set point of 2485 psig. The controlled lift of any SRV would then end the overpressure transient.

The potential then exists for pressurized thermal shock (PTS) to cause vessel failure. Such vessel failures are typically assumed to be catastrophic, leading directly to core-melt. Note that if PTS-induced vessel failure does not occur, no safety consequences are expected as a result of this scenario.

The NRC is currently funding a program at ORNL to examine PTS related issues in PWRs, including the H. B. Robinson plant as a representative Westinghouse plant. The results have not yet been published, but a review of applicable information is possible for inclusion in the A-47 program at this time. A similar overpressure type of scenario in the H. B. Robinson plant was considered, with overpressure to 2200 psig and instantaneous drop in temperature to various levels. Again, using a nil ductility transition temperature of 270°F as discussed in Chapter 4, vessel failure probabilities of 2E-02 were obtained for a vessel temperature of 150°F. This failure probability dropped to 4E-09 at 350°F.

As discussed in Chapter 4, the actual nil ductility transition temperature for H. B. Robinson is 130°F. ORNL expects vessel failure probabilities to drop several orders of magnitude if 130°F is used rather than 270°F. It will then be assumed that the vessel failure probability for this overpressure scenario postulated for the A-47 program is 2E-04, using the more realistic plant data.

A core-melt frequency can then be estimated by considering the initiating event frequency, a probability that the operator fails to manually terminate the overpressure, and the probability of PTS vessel failure. The probability of the operator to fail to terminate the scenario will be estimated at 0.1 here, given the consistent information available. An estimate of the core-melt frequency is then (1.5E-07/py)(0.1)(2E-04) = 3.0E-12/py.

Public risk will be assumed to be associated with release category 3 at 5.4E+06 man-rem per event, giving a public dose of (3.0E-12/py)(5.4E+06 man-rem) = 1.6E-05 man-rem/py.

Again, these frequencies are considered to be negligible. They will be included only for completeness.

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7.0 REACTOR COOLANT SYSTEM OVERPRESSURE SEQUENCE 2

This sequence involves failures of the safety injection electronics that causes a spurious safety injection (SI) initiation signal during plant startup.

7.1 INITIAL PLANT CONDITIONS FOR OVERPRESSURE SEQUENCE 2

The initial plant conditions assumed by INEL had the reactor at startup with the RCS temperature at 350°F and pressure at 265 psia. During startup, the PORVs and SRVs are disabled until a system temperature of approximately 470°F is reached. A spurious initiation of high pressure safety injection thus results in a pressure rise that exceeds the technical specification temperature-pressure limit.

The sequence of events postulated by INEL and the required time for each event to take place are shown in Table 7.1.

TABLE 7.1. Sequence of Events for Overpressure Sequence 2

Time (s)	Event
0	SI spurious signal
6	Accumulator isolation valves fully opened
10	High head SI discharge valves fully opened
13	Accumulator check valves closed
162	Pressure limit exceeded for existing RCS temperature
192	Calculation terminated

The accident initiator postulated by INEL thus involves no initial failure of the primary containment or cooling of the core.

7.2 ACCIDENT PROGRESSION ANALYSIS FOR OVERPRESSURE SEQUENCE 2

The Westinghouse PWR uses two centrifugal charging pumps for the high head SI function, with suction from the boron injection tank. The two centrifugal pumps provide 2 x 75 = 150 gpm at 2800 psig. (In addition, the charging system has one reciprocating pump not used for high head SI, which provides 98 gpm at 3200 psig.

The pumps from the Residual Heat Removal System (RHRS) are used for the dual purpose of low head SI as well as heat removal. These two pumps have a design pressure of 600 psig, with a design flow of 3000 gpm. Operation of the RHRS is typically initiated at pressures and temperatures of approximately 425 psig and 350°F, respectively.

The following reactor coolant system designs pressures apply:

Hydrostatic test pressure	3110 psig
Design pressure	2485 psig
Safety valves	2485 psig
PORVs	2335 psig

In the scenario defined by INEL, the plant is in startup mode, which implies that the PORVs are disabled until the system reaches 470°F. The system pressure exceeds that of the low head SI contribution from the RHRS pumps, as pointed out by INEL. Thus, only the centrifugal charging pumps are of importance.

It is assumed that both charging pumps are running, with a maximum pressure head of 2800 psig. Disabling the PORVs would then bring the system past the PORV lift pressure of 2335 psig, then past the safety valve set point of 2485 psig. The controlled lift of any SRV would then end the overpressure transient, with the vessel pressure oscillating about the SRV set point, which is likely to be in the 2200 to 2485 psig range.

As developed in the previous chapter, the potential then exists for pressurized thermal shock (PTS) to cause vessel failure, leading directly to core-melt.

Again, relying on information from the PTS program at ORNL, the vessel failure probability for an overpressurization at 2200 psig at 350°F was set at 4E-09, with a conservative nil ductility transition temperature of 270°F. As discussed in Chapter 4, the actual nil ductility transition temperature for the H. B. Robinson plant is 130°F. It will then be assumed that the vessel failure probability for this overpressure scenario postulated for the A-47 program is 4E-11, using the more realistic plant data. Probability estimates this low can essentially be considered insignificant, but again will be carried through to core-melt for completeness and comparison to the other risk contributors being considered in the A-47 program.

A core-melt frequency can then be estimated by considering the initiating event frequency, a probability that the operator fails to manually terminate the overpressure, and the probability of PTS vessel failure. The probability of the operator to fail to terminate the scenario will again be estimated at 0.1, as in the previous overpressure scenario, given the consistent information available. An estimate of the core-melt frequency is placed at (3.7E-04/py)(0.1)(4E-11) = 1.5E-15/py.

Public risk will again be assumed to be associated with release category 3, at 5.4E+06 man-rem/event, giving a public dose of (1.5E-15/py)(5.4E+06 man-rem) = 8.1E-09 man-rem/py.

Again, these frequencies are considered to be negligible. They are included only for completeness.

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8.0 STEAM GENERATOR TUBE RUPTURE SEQUENCE 1

This sequence involves opening a break in one steam generator tube adjacent to the cold leg tube sheet. Additionally, a complete loss of offsite power is also analyzed as an aggravating failure.

8.1 INITIAL PLANT CONDITIONS FOR STEAM GENERATOR TUBE RUPTURE SEQUENCE 1

The initial plant conditions for Steam Generator Tube Rupture Sequence 1 are a plant at 102 percent reactor power, the RCS pressure at 2280 psia, and all systems controlling in automatic. No system failure modes were identified by INEL for initiating a steam generator tube rupture. Therefore, it was assumed by INEL that this sequence is initiated by opening a break in one steam generator (SGA) tube adjacent to the cold leg tube sheet. A complete loss of offsite power was also analyzed by INEL as an aggravating failure.

The sequence of events postulated by INEL and the required time for each event to take place due to steam generator tube rupture are shown in Table 8.1.

TABLE 8.1. Sequence of Events for Steam Generator Tube Rupture Sequence 1

Time (s)	Event
0.0	Transient initiated by opening a break in one steam generator tube in SGA adjacent to the cold leg tube sheet. Additionally, a complete loss of offsite power was assumed.
	Turbine tripped
	Reactor tripped
	RCPs tripped
	MFW pumps tripped
	Condensate pumps tripped
	Pressurizer heaters tripped
1.0	Control rod banks begin insertion
	Turbine stop valves closed
3.6	All control rod banks fully inserted
7.5	MFW valves closed

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TABLE 8.1. (Cont'd)

Time (s)	Event
8.0	Steam dump valves closed for remainder of transient
10.0	Motor driven AFW system valves fully opened
21.2	Turbine driven AFW initiated on 2/3 low-low steam generator NR levels (15 percent)
65.0	SGA PORV pressure setpoints (1050 psia) reached. PORV sticks in the fully opened position
81.0	SGA secondary system pressure lowest of three SGs. All AFW preferentially flowed into SGA
86.9	SIAS initiated due to steam header pressure 100 psi higher than SGA steam line pressure
250.0	Pressurizer empty
360.0	Reactor vessel upper head began voiding
388.3	MSIVs closed on 2/3 low primary coolant loop temperatures [<543 ^{o}F]
589.0	57000 lbm released to atmosphere through stuck open SGA PORV
660.0	AFW sources to SGA manually isolated
780.0	Calculation terminated

Primary to secondary mass	= 46040	lbm
Secondary to atmosphere mass	= 69394	lbm
Average break flow rate	= 58.6	lbm/s
Average PORV flow rate (60 s)	= 63.5	lbm/s
Estimated time for break flow into SGA to reach 70000 lbm	= 1190	s
Estimated mass discharged to atmosphere by 1190 s	= 95429	lbm

A detailed discussion of the analysis done on accident progression to core-melt is provided in the following sections.

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8.2 ACCIDENT PROGRESSION ANALYSIS FOR STEAM GENERATOR TUBE RUPTURE SEQUENCE 1

The initiating events defined by INEL for this event already include the failure of some highly important systems, primarily the loss of offsite power. As a result, the bounding sequence leading to core-melt will be the failure of the emergency electrical power supplies needed for the core spray systems.

This is shown in Figure 8.1, including a simplified version of the response of safety injection systems to demonstrate that the emergency power branch does bound the problem when coupled with the failure data presented below. This will be used rather than the more extensive event tree from the Zion plant presented earlier.

Note that the steam generator tube break (rupture) has been analyzed in the ORNL Precursor Study (Minarick and Kukielka 1982) and also in the INPO review of the ORNL's Study (INPO 1982). Therefore, in both studies the initiating event is simply "steam generator tube rupture," with no system failure modes identified which would cause the rupture.

In the ORNL assessment, failure of safety injection was assumed to lead to core damage. However, INPO's assessment shows that isolation of the affected steam generator and successful operation of charging and isolation of letdown is sufficient in lieu of loss of safety injection.

One other possibility in this analysis is the RCS overcool due to stuck open steamline PORV. This scenario has already been analyzed and is of secondary concern compared to steam generator tube rupture. As has already been mentioned, the main concern is to stop the break flow to limit the release of radioactive coolant to the atmosphere. Therefore, no attempt has been made to analyze this aggravating side effect.

The specific steps of the tree are described below.

Initiating Event. The INEL initiating frequency of 2E-03/py considers only the loss of offsite power (LOOP) with PORV lift. Again, INEL did not identify any failure mechanism that would result in a SGTR given this event. The pressure transient is within design specifications of new SG tubes, and as such would normally not present a credible event to cause rupture. However, the potential certainly exists for degraded tubes to be present in a plant after operation has begun. As a result, two cases should be considered to estimate the frequency of a coincident SGTR with LOOP and PORV lift: the first would be a LOOP and PORV lift which in turn causes a SGTR, and the second a SGTR with LOOP and PORV lift occurring during recovery period following the SGTR. These will be developed below.

Case 1 will consider the probability of causing a SGTR following LOOP and PORV lift with a frequency of 2E-03/py. The probability of SGTR used previously was for SGTR following a MSLB, put at 0.034 for single and multiple tube rupture. The probability of a single tube rupture only was 0.017. In this case, the PORV lift could be expected to represent a less severe pressure transient than a full MSLB. It is proposed that a failure probability of (0.1)(0.034) + 0.0034 be used for single or multiple tube ruptures, and 0.0017

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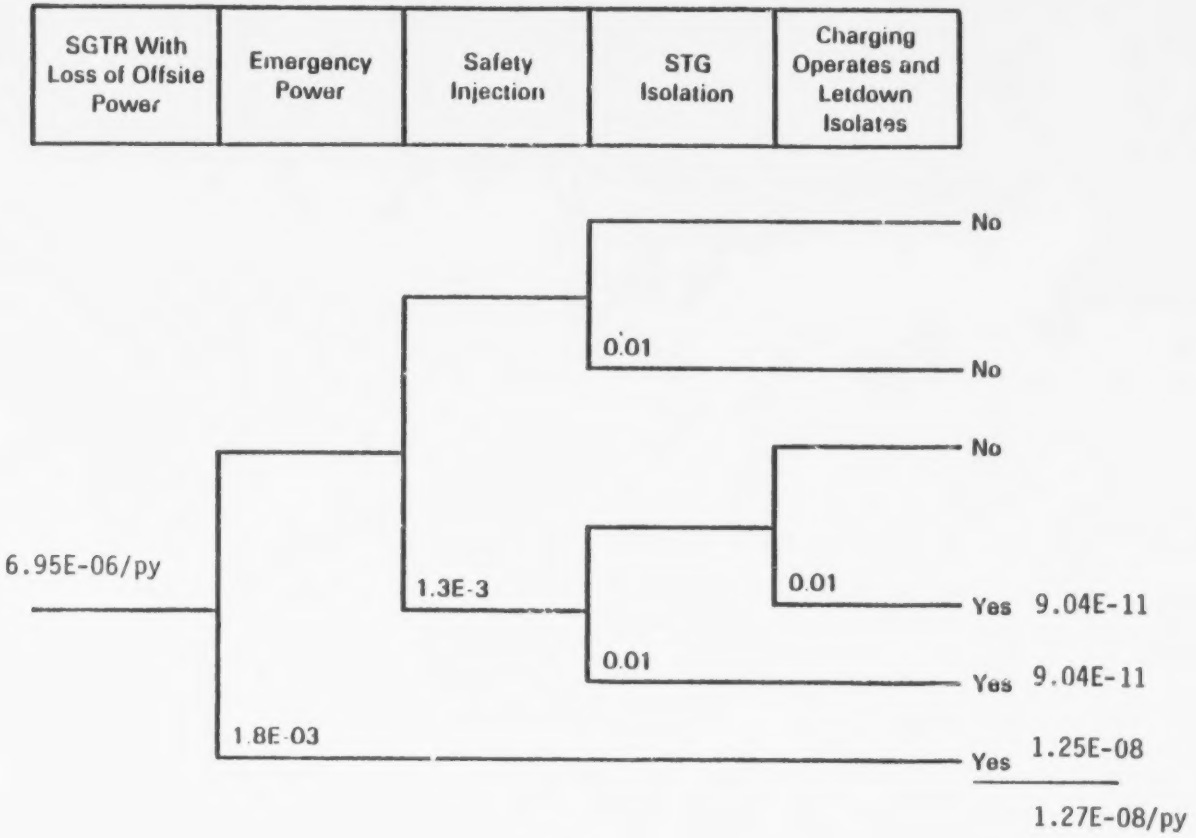

FIGURE 8.1. Steam Generator Tube Rupture Event Tree Sequence 1

for a single tube rupture following PORV lift. Although INEL modeled a single rupture, the 0.0034 value will be used here since it is more representative of the estimate of SGTR in general. The frequency of LOOP, PORV lift, and SGTR is then estimated to be (2E-03/py)(0.0034) = 6.8E-06/py.

For Case 2, the potential for a LOOP and PORV lift during a 24-hour period following SGTR will be considered. The frequency of random SGTRs was assumed to be 0.022/py in NUREG-0844 (U.S. NRC 1985). Using the INEL value of 7008 hr/py, the frequency of SGTR followed by LOOP and PORV lift during a 24-hour recovery period is then (0.022 SGTR/py)(2E-03 LOOP with PORV lift/py)(1 py/7008 hr) (24 hr) = 1.5E-07/py.

The sum of both cases is then 6.8E-06/py + 1.5E-07/py = 6.95E-06/py. This will be used as an estimate of the frequency of this event driven by the control system failures of interest.

Emergency Power. Since a complete loss of offsite power is also assumed in this scenario, the startup of the emergency power is required for operation of safety systems. The failure probability of emergency power to start is assumed to be 1.8E-03. This estimate is obtained from the ORNL study (Minarick and Kukielka 1982). It is assumed that steam generator tube rupture does not affect the functionability of the emergency power.

Safety Injections. The failure probability of the safety injection system is assumed by both the ORNL and the INPO study (INPO 1982) to be 1.3E-03. Failure of safety injection to operate during a tube rupture event would not necessarily lead to core damage. Since the affected steam generator can be isolated, the loss of primary coolant can be curtailed. With steam generator isolation, proper operation of charging and letdown would prevent core damage.

Ruptured Steam Generator Isolation. This step requires both closure of the isolation valve to the affected generator, and isolation of the feedwater flow. The probability of failure to isolate the affected steam generator was estimated to be 0.1 by the ORNL study. INPO has, however, argued that isolation of the affected steam generator is a step in the tube rupture procedure and should be accomplished with a higher reliability than 0.9. For a well-recognized procedural step, a reliability of 0.99 is suggested by INPO and will be used here.

Charging Operates and Letdown Isolates. The proper operation of charging and letdown, combined with seat generator isolation, can prevent core damage even if the safety injection system has failed. The failure probability to properly operate charging and letdown is estimated by INPO to be 0.01. This estimate is used in this analysis.

8.3 FREQUENCY OF CORE-MELT DUE TO TUBE RUPTURE SEQUENCE 1

The predicted frequency of core damage due to failing open of the steamline relief valve is estimated to be 1.27E-08/py, with over 98% of this due simply to failure of emergency electric power. If no corrections are applied to distinguish between core damage and core-melt, the predicted frequency of core-melt is then 1.27E-08/py. As can be seen, the failure of emergency electric power dominates this probability.

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8.4 PUBLIC RISK DUE TO TUBE RUPTURE SEQUENCE 1

Given the total failure of the electric power supply, the worse case release category 1 will simply be assumed, with highest associated public dose of 5.4E+06 man-rem/core-melt. The resulting public dose is then (1.27E-08/py)(5.4E+06 man-rem) = 6.86E-02 man-rem/py.

8.5 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR SGTR SEQUENCE 1

The total core-melt frequency predicted as a result of this issue is again 1.27E-08/py.

The total predicted dose associated with this issue is then 6.86E-02, or 6.9E-02 man-rem/py.

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9.0 STEAM GENERATOR TUBE RUPTURE SEQUENCE 2

The initiating failure for this transient is similar to that of SGTR Sequence 1. A tube rupture is assumed to occur to initiate the event. As before, no system failures could be identified by INEL that could cause a tube rupture. The aggravating system failures assumed for this sequence are a failure that results in an increase in feedwater flow to the steam generator with the rupture and a failure that results in the affected steam line PORV sticking open when it initially opens to relieve the increasing steam generator pressure.

9.1 INITIAL PLANT CONDITIONS FOR STEAM GENERATOR TUBE RUPTURE SEQUENCE 2

The initial plant conditions for SGTR Sequence 2 are the same as in the previous scenario. It is assumed that the plant is at 102 percent reactor power, the RCS pressure at 2280 psia, and all control systems are in automatic. A tube rupture is assumed to initiate this transient. The aggravating failures considered include an increase in feedwater flow to the affected steam generator, and a stuck open steam line PORV.

The INEL analysis showed that at 60 seconds the break flow rate was 89.5 lbm/s and the feedwater flow rate into the affected steam generator was 1116.8 lbm/s, which results in a total flow of 1206.3 lbm/s into the steam generator. The only mass being removed from the SG is due to steam flow, which was calculated at 996.8 lbm/s. Therefore, the net effect on the SG is an increase in mass of 209.5 lbm/s, which corresponds to a rate of increase of 0.4%/s in the steam generator water level. The analysis for SGTR Sequence 1 showed that the PORV opening at 65 seconds after loss of offsite power would result in an immediate reactor trip and turbine trip. For SGTR Sequence 2 the turbine trips occur at 61 seconds from high SG level and the reactor trip would be actuated from the turbine trip. The exact time of the PORV actuation could not be determined by INEL, but it was postulated to occur after water enters the steamline at 123 seconds.

The remainder of this sequence is expected to be similar to SGTR Sequence 1. A more detailed discussion of the analysis of accident progression to core-melt follows.

9.2 ACCIDENT PROGRESSION ANALYSIS FOR STEAM GENERATOR TUBE RUPTURE SEQUENCE 2

In a manner similar to the previous steam generator tube rupture scenario, several aggravating failures were considered here. These were increase in feedwater flow to the steam generator and stuck open steam line PORV. There are no direct common mode failures for these two modes; however, there is a possibility that the steam generator overfill aggravated by the high feedwater flow rate could result in two phase flow or a water slug being discharged out of a steamline safety valve. This could cause chattering or valve damage, reducing the reliability of the valve to reseat. Again, as developed in the BWR

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analysis, the available information indicates that chattering may occur, but no reduction in valve reliability has been detected. Standard accepted valves for PORV performance will therefore be used.

Initiating Event. As in the previous chapter, the INEL initiating frequency of 3.2E-03/py considers only the feedwater increase with PORV lift. INEL did not identify any failure mechanism that would result in a SGTR given this event. However, the potential again exists for degraded tubes to be present in a plant after operation has begun. As a result, two cases will again be considered to estimate the frequency of a coincident feedwater increase with PORV lift and SGTR: the first would be a feedwater increase and PORV lift, which in turn causes an SGTR, and the second would be an SGTR, with feedwater increase and PORV lift occurring during the recovery period following the SGTR. These will be developed below.

Case 1 will consider the probability of causing a SGTR following feedwater increase and PORV lift with a frequency of 3.2E-03/py. The 0.0034 value will be used here as more representative of the estimate of SGTR in general following PORV lift, again a factor of 10 less than the 0.034 used in previous chapters for SGTR following a MSLB. The frequency of feedwater increase, PORV lift, and SGTR is then estimated to be (3.2E-03/py)(0.0034) = 1.09E-05/py.

For Case 2, the potential for a feedwater increase and PORV lift during a 24-hour period following SGTR will be considered. The frequency of random SGTRs was assumed to be 0.022/py in NUREG-0844. Using the INEL value of 7008 hr/py, the frequency of SGTR followed by feedwater increase and PORV lift during a 24-hour recovery period is then (0.022 SGTR/py)(3.2E-03 LOOP with PORV lift/py)(1 py/7008 hr)(24 hr) = 2.41E-07/py.

The sum of both cases is then (1.09E-05/py + 2.41E-07/py) = 1.11E-05/py. This will be used as an estimate of the frequency of this event driven by the control system failures of interest.

Note that the INEL estimate is based on a feedwater failure contribution of 8.8E-06/hr and a valve lift contribution of 5.2E-02/hr over a period of 7008 hours, or (7008 hr/py)(5.2E-02)(8.8E-06)/hr = 3.2E-03/py. The valve failure frequency includes a factor for failure of 2E-02/demand, thus dominating the contribution fo PORV or relief valve lift. This, however, assumes that a demand signal for valve lift will be generated with the feedwater increase, with the PORV then sticking open. This assumption may have been appropriate for the SGTR Sequence 1, where a LOOP would most likely result in a PORV lift due to rising pressure. It is highly uncertain if such is the case for a scenario initiated by feedwater increase, as with Case 1. It is thought that an active control failure, likely 1E-06/hr, would be required here for coincident PORV lift. For Case 2, however, initated by an SGTR, a PORV lift could be expected. As a result, the Case 1 frequency of 1.09E-05/py is thought to actually be much lower. The Case 2 frequency of 2.41E-07/py is then thought to be a more correct estimate of the initiating frequency for SGTR Sequence 2.

This uncertainty is noted, but the original INEL frequencies will be used to estimate risk.

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9.3 FREQUENCY OF CORE-MELT DUE TO STEAM GENERATOR TUBE RUPTURE SEQUENCE 2

This scenario introduces a number of control system failures that aggravate the cooldown rate on the secondary side. The increased feedwater flow and steam line PORV lift add to the cooldown transient. The additional failures do not in themselves impact the performance levels of systems that are required to respond to the scenario. However, the aggravating cooldown may make the required response different than that required for a SGTR alone. The primary concern is that the combination of loss of inventory plus shrinkage due to cooldown will create a more severe accident.

In this scenario, the feedwater increase can again lead to overfill and spillover into the steam lines. However, as developed in Chapters 2 and 3 for the overfill sequences, the progression to SGTR and the actions required to recover from SGTR are seen to dominate the core-melt and risk contributions.

For this scenario, a steam line break already exists in the form of a PORV or steam dump valve lift. In addition, a coincident SGTR was also assumed. As a result, the same considerations that were developed in Chapter 2 section 2.5 again apply here. It was shown that recovery from a SGTR with MSLB depended greatly on the ability to isolate the affected steam generator. Conditional probabilities of core-melt were then developed for MSLB both above and below the MSIVs.

In this case, however, the steam line break is due to a PORV or steam dump valve lift, both of which can be isolated. The failure probabilites of interest needed for recovery in this scenario were developed in Section 2.5 as Case 2: Rupture of Main Steam Line Downstream of the MSIV (i.e., an isolatable break).

As INEL postulated, the probability of loss of reactor storage water before RCS depressurization for a single SGTR was put at 1E-04, and the probability of failure to isolate the steam generator was put at 1E-03. The conditional probability of core-melt given the overfill, steam line valve lift and SGTR would then be (1E-04)(1E-03) = 1E-07.

The estimated core-melt frequency for this sequence would then be the initiating frequency times the conditional probability of core-melt, or (1.11E-05/py)(1E-07) = 1.11E-12/py.

The risk will be estimated by assuming release category 2 as in Chapters 2 and 3, giving (1.11E-12/py)(4.8E+06 man-rem/event) = 5.33E-06 man-rem/py.

9.4 TOTAL CORE-MELT FREQUENCY AND PUBLIC DOSE FOR STEAM GENERATOR TUBE RUPTURE SEQUENCE 1

The total core-melt frequency predicted as a result of this issue is again 1.11E-12/py.

The total predicted dose associated with this issue is then 5.33E-06 man-rem/py.

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10.0 VALUE/IMPACT ANALYSIS OF POTENTIAL CORRECTIVE FEATURES

In this chapter, various modifications will be postulated to correct the control system failures identified by INEL. An estimate will be made of the effectiveness of such fixes in reducing or eliminating the failure frequencies, and this will be translated into effective reductions in core-melt frequency and public risk. The cost of implementing such corrective features will also be estimated. The purpose of this is to provide a range of value/impact ratios of man-rem saved per $1000 associated with the various failure modes and fixes identified for the A-47 issue.

Again, the sequences of interest are as follows:

Overfill Sequence 1 - False MFW Increase, Trip, and AFW Overfill
Overfill Sequence 2 - False MFW Increase and High Level Trip Failure
Overcool Sequence 1 - Inadvertent Opening of Steam Dump Valves at Power
Overcool Sequence 2 - Inadvertent Opening of Valves During Hot Shutdown
Overpressure Sequence 1 - Loss of Letdown Flow and PORV Failure at Shutdown
Overpressure Sequence 2 - Inadvertent Safety Injection at Shutdown
Steam Tube Rupture Sequence 1 - SGTR with LOSP and Stuck Open PORV
Steam Tube Rupture Sequence 2 - SGTR with High MFW Flow

These will be discussed below.

10.1 OVERFILL SEQUENCE 1 - FALSE MFW INCREASE, TRIP, AND AFW OVERFILL

In Section 4.6 of the INEL report, the failures of interest are listed as a failure of the controlling level transmitter, a pipe leak or rupture on the controlling instrument line, inadvertent opening of an air operated valve for the MFW, and failure of a circuit in the feedwater level controller and failures in the steam flow or feedwater flow instruments. The failure data are reproduced below.

- level control instrument failure	1E-06/hr
- leak in line	5E-09/hr
- regulating valve failure	3E-07/hr
- level control circuitry failure	1E-06/hr
- feedwater flow instrument failure	1E-06/hr
- steam flow instrument failure	1E-06/hr
	4.3E-06/hr

This is assumed to convert to the INEL initiating frequency of 1.4E-03/py. As shown, individual modifications to the plant to reduce the 2.0E-03/py initiating frequency of this event will only impact small fractions of the overall initiating frequency. Individual modifications are discussed below.

Postulated Fix: Changes in the Level Control High Level Trip Logic

In this sequence, the feedwater increase can be driven by a feedwater level controller failure, as noted above. However, the high-level trip functions as

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designed, with overfill continuing with the auxiliary feedwater system. As a result, the addition of another level transmitter or modifications to the trip logic will have no impact on risk reduction for this sequence. The value/impact of such modifications would then be zero for this scenario.

This will be of interest for Overfill Sequence 2.

Postulated Fix: Better Weld Integrity on Instrumentation Lines

Note that this fix would impact Overfill Sequences 1 and 2. In Sequence 2 the leaking line generates a low-level indication and fails the high level trip. The low initiating frequency of Sequence 2 makes this an insignificant contribution, however.

As with the BWR, the weld points on the 2-inch instrumentation lines were assumed by INEL to be the weak points subject to leakage or rupture. Better QA of welds (i.e., radiography or other nondestructive evaluation [NDE]) could reduce the postulated failure rate. The lines may not be subject to the same stress corrosion cracking problems as in BWRs, but this will be assumed here, which would thus require annual inspection to be effective.

The reduction in weld failure is likely to be small, given the QA to which the pipes are now subjected. A reduction in weld failures of 10 percent will be estimated here, as was done for the BWR. The reduction in frequency is then (0.1)(5E-09/4.3E-06) = 1.16E-04, or a 0.01% reduction in the initiating frequency.

The reduction in core-melt frequency is then (1.16E-04)(6.17E-08/py) = 7.12E-12/py. The reduction in risk is then (1.16E-04)(3.0E-01 man-rem/py) = 3.5E-05 man-rem/py, or 1.0E-03 man-rem over 30 years.

NUREG-1061 (U.S. NRC 1984) puts the cost of NDE piping inspection at approximately $3000/weld inspected. For such a small line that is not safety grade, much of the cost associated with QA will not be applicable. This cost is reduced here to $500/weld, divided between labor and QA/records costs. The annual outage cost for 12 welds per instrument line and 2 instrument lines is then $12,000/py. At a 10 percent assumed discount rate over 30 years, this represents a cost of ($12,000)(9.43) = $1.13E+05.

The value/impact ratio is then

(1.0E-03 man-rem)/($1.13E+05) = 8.8E-06 man-rem/$1000.

As with the BWR, note further that NUREG-1061 also predicts an occupational exposure of 0.8 man-rem per weld inspected for the large pipes. The instrument lines also carry reactor coolant, but the radiation field would be expected to be significantly lower around the small pipe and away from the reactor vessel, being located on the steam generators in the PWR.

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Postulated Fix: Hardened Instrumentation Lines

Note that this would impact Overfill Sequences 1 and 2. In Sequence 2 the leaking line generated a low level indication and failed the high level trip. As before, however, the low initiating frequency of Sequence 2 makes this contribution insignificant.

The pipe runs (likely 304 stainless steel) can be changed using a material like 316 SS, which is more resistant to stress corrosion cracking. The 316 stainless steel piping would be expected to significantly reduce the frequency of weld failures due to the reduction in stress corrosion cracking. It is estimated that the frequency of pipe weld leakage could be reduced by 75 percent. It is uncertain that any reduction in pipe rupture frequency would be achieved.

Note that from the data above, leakage only contributes a fraction of 5E-09/4.3E-06 = 0.001 to the initiating frequency. An estimated fraction of reduction in the initiating frequency is then (0.75)(5E-09/4.3E-06) = 8.72E-04. The estimated reduction in core-melt frequency is then (8.72E-04)(6.17E-08/py) = 5.38E-11/py. The reduction in dose is then (8.72E-04)(3.0E-01 man-rem/py) = 2.62E-04 man-rem/py, or 7.8E-03 man-rem over 30 years.

The cost for replacing the piping, replumbing and recalibrating all instruments is estimated at 4 man-weeks per instrument line or 8 man-weeks of installation labor. The engineering and QA time is put at 4 man-weeks, for a total of 12 man-weeks at $2270/man-week, or $27,240. The material costs are put at $5000 for approximately 200 feet of piping and fittings, for a total cost of $32,240.

- 4 man-weeks of engineering support at $2270/week, or $9080
- 8 man-week of craft services at $2270/week, or $9080
- $5000 in instrumentation and supplies.

The value/impact ratio is then

(7.8E-03 man-rem)/($32.2E+03) = 2.4E-04 man-rem/$1000.

Postulated Fix: Automatic Shutoff of the Auxiliary Feedwater

The actual overfill in this case is driven by the AFW system. The addition of a high water level trip to the AFW electric pumps would terminate the scenario. The high water level switches already in place could serve this function. If higher water levels are routinely allowed during shutdown, a pressure-enable could be added to the logic, requiring pressure to be above some level (i.e., 300 psig).

This would essentially eliminate the sequence, resulting in a core-melt reduction of 6.17E-08/py, and a public dose reduction of 3.0E-01 man-rem/py, or 9.0 man-rem over 30 years.

Development Cost

No NRC generic issue evaluation costs are foreseen, but each utility would likely fund studies on the impact of modifying this control system. The cost of these studies is estimated to be:

2 man-months of utility time, or (8)($2270) = $18,160.

Implementation Cost

Assuming that this fix is implemented during normal outages, it is estimated that the actual work will require approximately the following:

- 2 man-weeks of engineering support at $2270/week, or $4540
- 1 man-week of craft services at $2270/week, or $2270
- $2000 in instrumentation and supplies.

This comes to $8810. Once implemented, no recurring costs for the utility or the NRC are foreseen. The total is then put at $18,160 + $8810 = $26,970.

The estimated value/impact ratio is then 9.0 man-rem/$26,970 = 0.3 man-rem/$1000.

Postulated Fix: Modification to Air Operated Feedwater Valve

The final modification that will be considered here for Sequence 1 is the modification of the air-operated control valve, with the failure frequency assumed by INEL to be approximately 3E-07/hr by INEL. Note, however, that this failure frequency is that for a normal air-operated valve failing to remain open. In the case of the reactor feedwater valves, an air-operated-spring-to-open valve is used such that loss of all air pressure will result in the valve opening which could cause the feedwater increase. However, the valves also incorporate a lock-up feature into the air supply system such that loss of air pressure will lock the valve in its existing position at the time of failure. The valve failing open would then require failure of the air supply, and failure of the lock-up feature. The actual failure rate for the system as designed should then be less than that presented here.

The system then already has safeguards for valve air supply failure, along with the high level trips and back-up support of the operator. Modifications to the control circuit to prevent false full open signals may also make the control of the valve less reliable. Given the safeguards already present, it is not thought to be necessary to postulate and evaluate the effect of an additional automatic trip or control circuit for the feedwater pumps.

If a modification to the valve were postulated, the risk reduction could at most approach (3E-07/4.3E-06) = 7E-02 of the total risk due to this sequence, or (7E-02)(3E-01 man-rem/py) = 2.1E-02 man-rem/py or 0.63 man-rem over 30 years. The cost of modifications would have to then be left under approximately $1000 to achieve a value/impact ratio of approximately 1 man-rem/$1000. This low a cost is considered unrealistic for any nuclear modifications.

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10.2 OVERFILL SEQUENCE 2 - FALSE MFW INCREASE AND HIGH LEVEL TRIP FAILURE

The low initiating frequency given by INEL results in estimated core-melt frequencies below 1E-10/py for the sequence. This is considered negligible, and no credible modifications postulated could still obtain meaningful value/impact ratios.

Postulated Fix: Changes in the Level Control High Level Trip Logic

In this sequence, the feedwater increase can be driven by a feedwater level controller failure, with failure of the high level trip function. In the INEL BWR report this failure frequency was significant at 6.5E-03/py, but INEL has put a much lower frequency of 5.5E-08/py for a similar failure in the Westinghouse plant. As a result, the benefits associated with an improved trip logic will be insignificant in this case, but will be discussed further for completeness.

Going to a 2-out-of-4 rather than a 2-out-of-3 trip logic would require the addition of another level transmitter. In the PNL examination of such a modification for the BWR for this program, an INEL estimate of a 50 percent reduction in the total overfill initiation frequency was assumed. Examining the failure contributions above, the simple addition of one transimitter would only result in a reduction of (0.5)(1E-06/4.3E-06) = 0.23 of the original initiating frequency. This would imply that additional circuitry changes would accompany the addition of one transmitter if a 50 percent reduction is to be achieved. This will be assumed here, giving an estimated reduction in core-melt frequency of (0.5)(1.10E-11/py) = 5.50E-12/py. The reduction in public risk is put at (0.5)(5.2E-05 man-rem/py) = 2.60E-05 man-rem/py, or 7.80E-04 man-rem over an assumed plant life of 30 years.

The cost of this modification was estimated in the BWR report at $150 thousand to $1 million, depending on the need for additional penetrations. The PWR and BWR also differ in that instrument lines would extend to the steam generator shell rather than the vessel itself. The $150 thousand figure will be used here as representative of the PWR. The value/impact ratio is then (7.8E-04 man-rem)($150E+03) = 5.2E-06 man-rem/$1000.

10.3 OVERCOOL SEQUENCES 1 AND 2 - INADVERTENT OPENING OF STEAM DUMP VALVES AT POWER AND INADVERTENT OPENING OF VALVES DURING HOT SHUTDOWN

These two sequences involve an inadvertent opening of a steam relief valve during operation or hot shutdown, respectively. These were developed in Chapters 4 and 5. Note that PNL feels that this issue should apply only to the power operated relief valves (PORVs) upstream of the main steam isolation valves (MSIVs), and the steam dump valves downstream of the MSIVs. The code safety relief valves (SRVs) found upstream of the MSIVs are passive in nature only, relying on spring pressure with no powered operator. As such, they should not be considered as part of the A-47 issue.

As with Overfill Sequence 2, the low initiating frequencies and resulting core-melt and risk estimates give such small values that no credible

modifications could be proposed for Overcool Sequence 1 and arrive at meaningful value/impact ratios. The sequence does not seem to have a credible safety significance. As a result, the following discussion will apply to Overcool Sequence 2 only.

Referring to the fault tree of Overcool Sequence 2 as shown in Figure 4.3 of the INEL NUREG/CR-4326 report (Ransom et al. 1985), the following failure rates are used:

- PORV control circuit failure	1E-06/hr
- PORV mechanical failure	3E-07/hr
- PORV steam dump control circuit failure	1.3E-06/hr
- steamline safety valve fails open	1E-05/hr
- steam dump control and arming circuit failure	1.2E-11/hr
- steam dump valve mechanical failure	3E-07/hr
- turbine reset switch failure	0.5E-08/hr
TOTAL	1.29E-05/hr

This is assumed to correspond to the INEL calculated initiating frequency of 1.8E-02/py, based on 876 hr/py when the plant is in the necessary hot shutdown condition. As shown, individual modifications to the plant to reduce the 1.8E-02/py initiating frequency of this event will only impact small fractions of the overall initiating frequency. Note that the SRV failure listed above is the dominant failure contributor identified by INEL. Again, PNL feels that this should not have been included in the A-47 program, since the SRVs are passive spring-loaded valves without active controllers. Individual modifications are discussed below.

Two general modifications can then be proposed to reduce the frequency of the above scenarios. The most promising approach would be to make use of the block valves already installed in the system. Use of these valves would counter any type of premature or false actuation of the PORVs or relief valves. This also would not directly affect the control logic currently used for the PORVs. The second fix would be to try to reduce the frequency of false actuations due to valve logic controller failure. These fixes are discussed below, with modifications potentially reducing the frequency of both operational and hot shutdown failures of the valves.

Proposed Fix: Automatic Actuation of Isolation Block Valves

As discussed in Chapter 4, the PORVs and steam dump in question already have block valves in place capable of isolating flow to the failed valve. One fix is to install an automatic logic to close the upstream block valves if inadvertent opening or leakage is sensed. Automatic actuation of a block valve on leakage of a PORV is already the subject of Task Action Item II.K.3.2, Installation and Testing of Automated Power-Operated Relief Valve Isolation System, dealing with PORVs on the primary side. Note that some plants also have block valves on the steam dump valves that could also isolate the dump valve failure.

Negative impacts include the possible reduced flexibility of PORV operation in transient situations. It would likely be necessary to include an operator disable for the automated block valve function.

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In Chapters 4 and 5, the frequency of steam line break was assumed to be the initiating frequency times the probability of operator failure to close the block valve plus the probability of block valve failure, or (1.8E-02/py)(0.05 + 0.005) = 9.90E-04/py. Task Action Item II.K.3.2 puts the probability of failure of an automated block valve isolation circuit, rather than an operator, at 0.002, giving an initiation frequency of (1.8E-02/py)(0.002 + 0.005) = 1.26E-04/py. The net reduction in PORV failure is then (1 - 0.007/0.055) = 0.873 of the base case values. However, from the INEL failure data above, 1E-06 + 3E-06 = 2.6E-06/py; the PORV failure contributes only (2.6E-06/1.29E-05) = 0.2 of the total initiation frequency, including the steam dump valves, which gives 2.9E-06/1.29E-05, or again approximately 0.2. The risk reduction must reflect this, giving a fractional reduction of (0.873)(0.20) = 0.175. The reduction in core-melt frequency is then (0.175)(8.33E-07/py) = 1.46E-07/py. The reduction in risk is then (0.175)(3.9 man-rem/py) = 0.68 man-rem/py or 20.5 man-rem over 30 years.

Costs

The costs of modifying PORVs to safety grade were put in Issue 70 at $25,000 for each loop in the three-loop H. B. Robinson plant. Additional costs for safety analyses and license amendments brought the costs to approximately $250,000 per plant. However, this modification deals only with the block valve logic. For changes to the isolation logic only, costs were put as follows:

- 4 man-weeks at $2270/week for engineering support, or $9080
- 2 man-weeks at $2270/week for installation, or $4540
- $25,000 for miscellaneous electronics, logic relays, panels, wiring, etc.
- $25,000 for plant safety analysis but no license amendment.

This totals $63,620 per plant. No recurring annual costs or NRC costs are foreseen.

The value/impact ratio is then 2.05 man-rem/$63,620 = 0.32 man-rem/$1000. If the costs for identifying all steam dump valves are considered also, the risk remains the same, resulting in a significantly lower value/impact ratio.

Proposed Fix: Modifications to Valve Controller Logic

The failures identified by INEL also dealt with false PORV valve lift. Corrective features could be directed at reducing the frequency of such false signals. The exact failure mechanisms are uncertain at this time, but any changes in the control circuitry for valve operators is likely to require a more substantial cost, plus the potential for reduced reliability for valve operation.

As an estimate, it will be assumed that some type of additional enable will be required between the control logic and the valve operator. This fix will not eliminate failures of the valves themselves. As a result, the change in core-melt and risk is thought to be bounded by the reduction in false actuation signals which are thought to be a relatively small contribution of the failures listed above for the PORVs. The risk reduction for full elimination of this problem is put at (2.3E-06/1.29E-05) = 0.178. The core-melt frequency and risk would also be reduced by a factor of 0.178, or (0.178)(8.33E-07/py) = 1.48E-07/py for Overcool Sequence 2.

The public risk for full elimination of a false PORV actuation signal would be reduced by (0.178)(3.9 man-rem/py) = 0.69 man-rem/py, or 20.8 man-rem over 30 years.

Cost and Value/Impact Ratio

If costs are put at $63,620 per plant as they are for the fix to block valve logic, the value/impact ratio becomes 20.8 man-rem/$63.62E+03 = 0.327 man-rem/$1000. Modifications would not in fact result in the total elimination of false PORV actuation signals, so this value/impact ratio represents maximum possible risk reduction.

Safety Implications

Due to the role of power-operated relief valves in transient control, modifications to the relief valves will actually require safety studies and changes in the plant technical specifications. These costs are put at:

- $50,000 NRC generic issue evaluation (spread over 47 completed PWRs), or approximately $1000 per plant
- 6 man-months of utility time, or $54,500
- $4000 license amendment.

This gives an additional cost of $59,500, for a total cost of $63,620 + $59,500 = $123,120. This would give a value/impact ratio of 20.5 man-rem/$123.12E+03 = 0.17 man-rem/$1000 for the block valve modification, and 20.8 man-rem/$123.12E+03 = 0.17 man-rem/$1000 for the PORV logic modification. These values are thought to better reflect the actual costs involved, and will be carried through the final summary table for value/impact ratios.

10.4 REACTOR COOLANT SYSTEM OVERPRESSURE SEQUENCE 1

This transient is initiated when the RCS is water solid and in a low temperature and low pressure condition (cold shutdown). The failure mode of this scenario is a loss of letdown flow coupled with a failure of both pressurizer PORVs to open. The PORVs can fail either together or independently and their failure can occur any time prior to or during the time they are challenged by the increasing RCS pressure.

The following mechanisms that could cause this failure were identified by INEL:

1. Loss of power supply that feeds both a letdown valve and one of the PORVs, so that the letdown valve goes to its fail safe (closed) position and the PORV is rendered inoperable, and a single active failure of the second PORV caused by any of the following:

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- failure of the pressurizer pressure instrumentation
- failure of the PORV control circuitry
- failure of the valve
- failure in the mode switch so that the low-temperature, low-pressure mode is not selected
- failure of the power supply to the valve.

2. Independent failure of a letdown valve (closed) and of both pressurize PORVs, so that they do not open to relieve RCS pressure.

The median probability for this sequence is again put at 1.5E-07/yr. It is apparent that the current system design already requires multiple failures to block the function of both PORVs and the letdown valve. As a result, any fixes postulated will have a marginal return on plant safety.

The potential for vessel failure by PTS was considered possible, but the low initiating frequency resulted in core-melt frequencies below 1E-10/py. With the resulting insignificant risk, this makes the presentation of any value/impact ratio based on reduced plant core-melt frequency and risk difficult. However, several fixes will be discussed below for completeness.

Proposed Fix: Independent Power Sources to the Letdown Valve and PORVs

The loss of power supply to the letdown valve and PORV was identified by INEL as one of the dominant causes of this scenario's initiation. The failure rate of a power bus was put at 1E-08/hr for 876 hr/yr when the reactor is in cold shutdown, or 8.76E-06/yr. Providing independent power supplies to the letdown valve and PORV would decrease the frequency of common mode failures.

The most likely solution is to provide power from a safety grade bus to the PORVs. This reliability could be further improved by splitting the power to the two PORVs between the "A" and "B" safety grade bus power supplies. Note that due to the general design guidance not to mix safety and non-safety grade systems, this may also require an overall upgrade of the PORVs to safety grade, along with the necessary studies and technical specification changes. This type of upgrade is the subject of Safety Issue 70.

Given this modification, the assumption of independent power supplies is relatively certain. Of interest, then, is the question: Given the power failure in one system, what is the probability of failure in the second? A reasonable time interval must be assumed for this calculation, with 24 hours suggested as representative for repair to power circuits. Given an electrical power bus failure frequency of 8.67E-06/py, the new system would then have a failure rate of (8.67E-06/py)(8.67E-06/py)(1 py/365 days)(1 day) = 2.1E-13/py. This for all practical purposes would eliminate the potential loss of PORV or letdown valve failure due to power failure.

Costs

This modification would impact the safety grade electrical power supplies. As such, the minimum effort for safety studies is put at 6 man-months, or approximately $50,000. Actual engineering and plant modifications would be an

additional cost, estimated at $10,000 for both staff time and materials. License amendments would be put at approximately $4,000. This gives a total rough estimate of $64,000 per plant.

Postulated Fix: Modification of Low-Temperature, Low-Pressure Mode Switch

The failure of the mode switch to properly reconfigure the PORV control logic appears to be a possible common mode failure for both PORVs. This would require the switch to fail internally while indicating the proper condition. The logical modification is to include an indicator light for each switch logic position, giving a positive indication of circuit connection. Failure of the mode select function would then require both a switch failure and operator failure. If the circuit already has this feature, consideration of the role of the operator further reduces the likely frequency of this scenario. Additional modifications to the mode switch could then not be defended on a reliability basis.

Costs

Costs for such a simple modification are put at approximately $10,000 as a first estimate for engineering support, equipment and installation.

Postulated Fix: Modifications to the PORV Control

The single active failure of the PORV can be partially avoided by reducing the failure rates of the pressurizer pressure instrumentation and the PORV control circuitry. Note that the INEL report does not provide detailed information on this circuitry. However, the failures required to defeat both the PORVs and the letdown valve appear to be independent in nature. Given this, the increased complexity due to the addition of another layer of control circuitry to reduce failures cannot be justified from a reliability standpoint.

Costs

Modifications to the control circuitry or logic of the valve systems will require engineering support, materials and equipment, and installation by craft services. As a rough estimate, it is assumed that 4 man-weeks of engineering support are required for the development of any design changes, and that approximately 8 man-weeks of technical labor would be required. The cost of labor time is calculated based on $2270/man-week. This yields a total labor cost of $27,240. The cost of general supplies is estimated at $4000.

The total cost of implementing simple fixes is then put at $27,240 + $4000, or approximately $32,000. This cost is summarized below.

- Engineering support, 4 man-weeks at $2270/week
- Craft services, 8 man-weeks at $2270/week
- Miscellaneous supplies, $4000.

It is also assumed that the design changes will take place during normal fuel outage, so no power replacement cost has been considered in the cost analysis.

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Note again that modifications to the PORV logic could be interpreted as requiring extensive safety analysis and modifications to the plant Technical Specifications, as with the use of a safety grade power supply. The safety costs alone could easily exceed 6 man-months of utility time, or approximately $50,000. The NRC would likely also study the modifications. NRC costs of $50,000 would average several thousand dollars per reactor when spread over the industry.

10.5 REACTOR COOLANT SYSTEM OVERPRESSURE SEQUENCE 2

This transient is initiated when the reactor is being heated up from cold shutdown. It is postulated by INEL that during heatup procedures the pressurizer PORVs are shifted from the "low temperature" to the "normal" position and the ECCS systems are enabled. The failure mode that causes the transient is an inadvertent SI initiation. The following mechanisms by which this failure mode occurs were also identified by INEL.

1. a single logic circuit failure results in actuation of the safeguards sequence

2. independent failures that initiate high head safety injection flow and opening of the accumulator isolation valve

3. a single failure in one of the two safety injection actuation buttons that results in actuation of the safeguards sequence.

The failure frequency for this sequence is again put at 3.7E-04/yr.

Note that as with the previous sequence, the PNL analysis determined that although the overpressure exceeded current technical specifications in the INEL analysis, the potential for vessel failure due to PTS would be extremely low at 350^{o}F based on work at ORNL for the PTS program. The resulting estimated core-melt frequencies were again below 1E-10/yr, which is insignificant. This again makes the presentation of any value/impact ratio based on reduced plant core-melt frequency and risk insignificant. However, several fixes will be discussed below for completeness.

Postulated Fixes: Logic Circuit Modification

Although individual failure rate estimates are again not reported in the INEL study, it appears that a single failure in the SI injection logic would likaly be one of the leading causes of false SI injection. Modification to the circuit to place an additional enable is possible. However, this is done at the possible expense of reducing reliability of SI operation when actually demanded. The mode switch selector is also obviously designed to avoid enabling the SI system until the proper conditions in the vessel are achieved.

The problem appears to arise over a short interval during startup, when the output pressure and capacity of the high pressure SI pumps can exceed the allowable pressurization rate set by the relatively low temperature and pressure of the vessel. One possible solution is to delay enabling of the SI function. In this fashion, the SI initiation logic is not changed during normal operation, but only during the startup process.

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Another solution would be to provide a higher minimum pressure enable for the high pressure system during the startup process. The logic can be designed with a one-way enable such that once the new minimum pressure has been reached, a lowering of pressure would not negate the SI enable. A manual reset would then be required.

Costs

Modifications to the SI logic would impact a safety system, requiring significant safety analysis and license amendments. Utility efforts would likely not be less than 6 man-months, or $50,000 plus $4,000 for the amendment. Engineering design and craft services support plus materials would likely add costs similar to those estimated for the PORVs above, approximately $30,000. The total cost could then exceed $80,000 per plant for a modification which does not appear at this time to have any quantifiable safety significance.

Proposed Fix: Modifications to the Manual SI Actuation Button

The other single failure mode identified by INEL was the failure of either of two manual actuation buttons. The simplest modification would be to replace each of two single buttons with two buttons in series. This would then require both buttons of either pair to activate the system.

However, the system is now designed with two switches in parallel, either of which is capable of initiating the SI function. This is to insure that a single switch failure would not prevent manual operation. Changing the switch logic would thus increase the potential for failure of the logic. In addition, failures of buttons to make contact on demand are presumably more likely than random failures producing active connections.

Any changes in the logic to series buttons must thus presume that inadvertent actuation presented a greater safety hazard than failure on demand, which has not been shown to be the case. Such modifications are thus not recommended. Fixes to the SI actuation buttons would have to be limited to changes in quality, presuming that buttons less prone to random internal failures can be found and utilized.

10.6 STEAM GENERATOR TUBE RUPTURE SEQUENCES 1 AND 2

The plant status for the INEL evaluations puts reactor power at 102 percent, RCS pressure at 2280 psia, and all control systems in automatic. Sequence 1 assumed a loss of offsite power with stuck open steam side PORV, while Sequence 2 assumed a main feedwater increase to the affected steam generator along with a stuck open PORV.

No control system failures were identified by INEL for initiation of these sequences. Rather, steam generator tube ruptures were simply postulated with aggravating failures. PNL, however, also considered the case in which an initial PORV or relief valve lift could cause an SGTR event. The assumed initiating events were not demonstrated to be a cause for later control system failures. As a result, the scenarios consisted of a series of otherwise

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independent events. This is reflected in the low estimated frequency of the scenarios, again put at approximately 2E-03/py for Sequence 1 and 3.2E-05/py for Sequence 2 without SGTR, and 7.0E-06/py and 1.1E-05/py with SGTR for Sequences 1 and 2, respectively.

Note that with the SGTR as the initiator for these sequences, fixes associated with reducing the frequency of tube ruptures would be the most likely consideration. The whole question of tube integrity, however, is the subject of specific investigations at the NRC. The scenarios modeled by INEL do not appear to indicate where any control system modifications could be used to reduce the frequency of the initiating event, or where the SGTR contributes to later control system failures.

SGTR Sequence 1

For SGTR Sequence 1, the scenario was assumed by PNL to consist of a loss of offsite power (LOOP), resulting in PORV lift and SGTR, or an independent SGTR followed by a LOOP and PORV lift during the shutdown process. In either case, the dominant failure mechanism from a risk perspective is the LOOP which requires operation of emergency power or recovery of offsite power to prevent core-melt. Plant modifications directed at reducing the frequency of SGTR or LOOP, or at improving the recovery of electric power are considered to be better dealt with by specific NRC programs outside the A-47 program and thus will not be addressed further here.

Note that modifications to the PORV do not help in plant recovery from a risk perspective, because recovery is still being dominated by the LOOP and need for electric power. However, elimination of the PORV lift through the addition of an automatic actuation of the block valve, as discussed earlier, would eliminate 20 percent of the valve failures defined by INEL and given earlier.

Referring to Chapter 8, this would result in a reduction in core-melt frequency of (0.2)(1.27E-08/py) = 2.54E-09/py, and a risk reduction of (0.2)(6.9E-02 man-rem/py) = 1.38E-02 man-rem/py or 0.4 man-rem over 30 years. With the cost for such a modification put at $63,620, the value/impact ratio would be put at (0.4 man-rem)($63.62E+03) = 6.3E-03 man-rem/$1000.

Note that PORV modifications were discussed earlier for Overcool Sequence 2, with a risk reduction put at 20.5 man-rem. The value/impact of PORV modifications should also include the contribution from SGTR Sequence 2 to risk reduction, but at 0.4 man-rem, this is approximately a factor of 50 less than that from the Overcool Sequence 2 alone. The value/impact ratio considering the contribution from both sequences together thus would not change the value previously calculated for the Overcool Sequence 2, and this value/impact ratio will be carried through to the final summary tables.

SGTR Sequence 2

This scenario involves again a feedwater increase with a PORV or steam valve failing open, coincident with an SGTR. The feedwater increase and PORV failure frequency was put by the INEL fault tree as (7008 hr/py)(feedwater failure and valve failure) = (7008 hr/py)(8.8E-05)(5.2E-02)hr = 3.2E-03/py.

With coincident SGTR, the overall frequency was further reduced by PNL to 1.1E-05/py. Again, the probability of failure of recovery from a single SGTR with an isolatable break was quite small, giving a core-melt frequency of 1.11E-12/py and a risk of 5.33E-06 man-rem/py.

Modifications to reduce the frequency of overfill or steam valve lift, as discussed earlier, will also impact the frequency of this sequence. Note, however, that with the low core-melt frequency and risk estimated for this sequence, any further reduction in risk attributable to such modifications would be insignificant.

The value/impact ratios calculated earlier will therefore not change significantly as a result of considerations for the SGTR Sequences 1 and 2.

10.7 VALUE/IMPACT SUMMARY

Table 10.1 presents a summary of the recommended fixes along with their value-impact ratios. The value-impact ratio is the ratio of the public risk reduction (man-rem) to the cost of implementing the design change ($).

TABLE 10.1. Summary of the Value-Impact Analysis of the Proposed Fixes

Proposed Fix	Scenarios Affected	Estimated Cost ($)	Estimated Risk Reduction (man-rem)[a]	V/I Ratio (man-rem/$1000)[a]
Better Weld Integrity	SG Overfill Sequences 1 & 2	1.13E+05	1.0E-03	8.8E-06
Hardened Instrumentation Lines	SG Overfill Sequences 1 & 2	3.22E+04	7.8E-03	2.4E-04
Automatic Shutoff of the Auxiliary Feedwater	SG Overfill Sequence 1	2.69E+04	9.0	0.3
New Level Transmitter with 2-out-of-4 Trip Logic	SG Overfill Sequence 2	1.5E+05	7.8E-04	5.2E-06
Automatic Actuation of Isolation Block Valves	RCS Overcool Sequence 1 & 2	1.23E+05	20.5	0.17

TABLE 10.1. (Cont'd.)

Proposed Fix	Scenarios Affected	Estimated Cost ($)	Estimated Risk Reduction (man-rem)[a]	V/I Ratio (man-rem/$1000)[a]
Modifications to Valve Controller Logic	RCS Overcool Sequence 1 &2	1.23E+05	20.8	0.17
Independent Power Source to the Letdown Valve and PORVs	RCS Over-Pressure Sequence 1	6.40E+04	Negligible	Negligible
Modifications of LTOP Mode Switch	RCS Over-Pressure Sequence 1	1.00E+04	Negligible	Negligible
Modifications to the PORV Control	RCS Over-Pressure Sequence 1	3.20E+04	Negligible	Negligible
Logic Circuit Modification	RCS Over-Pressure Sequence 2	8.00E+04	Negligible	Negligible

(a) "Negligible" is indicated for scenarios with core-melt frequency estimates less than 1E-10/py.

As shown, the addition of a high level trip on the auxiliary feedwater is thought to be the most cost-effective measure for Overfill Sequence 1. The main feedwater trips function as designed in this scenario, so modifications to the main feedwater control logic are not called for.

Overfill Sequence 2, however, does combine a false low-level reading and loss of high trip function associated with the level transmitters and switches of the main feedwater system. An additional level transmitter and modification to the control logic for a 2-out-of-4 trip signal appear to reduce the frequency of such failures without a corresponding increase in false trip signals. However, INEL estimated that initiation frequency for this sequence is too low to justify any modifications to the trip logic.

Modifications to the instrumentation piping to reduce leakage and false low-level signals appear to provide a poorer value/impact for the two overfill scenarios than for the two modifications mentioned above.

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For the overcool scenarios, the addition of an automated block valve or modifications to the PORV logic appear to provide the most likely solution. Given the fact that the valves are already in place and that many plants are instrumented for tail-pipe steam flow, the cost of adding a trip would be minimal. Note, however, that modifications to the performance of the PORVs will impact directly the use of such valves in transient overpower scenarios. Thus it may be necessary to fund safety studies for block valve isolation, with changes to the plant procedures and technical specifications.

For RCS Overpressure Sequences 1 and 2, the estimated core-melt frequency was extremely low due to the low initiating frequency and probability of vessel failure due to PTS.

It is believed that due to the low frequency of initiation of these two scenarios, no more fixes are needed.

Note that the core-melt frequency and public risk for all of the above scenarios represent a best engineering estimate. The costs likewise represent a best engineering estimate; thus a certain amount of latitude is needed in interpreting the value/impact ratio. However, the development of these accident initiators to core-melt is thought to reflect a conservative approach to estimating the impact of these failures on plant engineered safety systems. Cost estimates likewise tend to underestimate the true cost of nuclear plant modifications. When combined, these factors tend to reduce the estimated value/impact ratios given above. The best estimates given above thus are thought to reflect the relative importance of each accident initiator and proposed fix, and its overall importance to plant safety.

All values for core-melt and public risk are so low, however, that the resulting low value/impact ratios make it difficult to mandate plant modifications for the A-47 program on the basis of these considerations.

11.0 CONCLUSIONS

The results of the analysis presented in this report are shown in Table 11.1. This table presents the frequency of postulated INEL control system failures and further estimates of the additional probability of progressing to core-melt. The associated public risk estimates are also given, as well as the upper bounds for the sequence initiating frequency. These frequencies were then propagated through appropriate event trees to core-melt, with the results presented below as a best estimate and upper bound.

The overfill and overcool scenarios postulated by INEL were primarily modeled as steam line breaks. The WASH-1400 study did not consider an overcool transient due to secondary side rupture as a viable initiator for a core-melt sequence. However, the steam line break is considered a precursor to core damage. Appropriate event trees from the ORNL and INPO study of precursors were then used, and the resulting frequency of core damage was very conservatively associated directly with core-melt. The public doses were then estimated by using release categories associated with core-melt sequences involving loss of high pressure injection and long term decay heat removal (release categories 3, 5, and 7).

In addition, the potential for inducing a steam generator tube rupture (SGTR) was considered. This probability was put at 0.034 given steam line break based on NRC considerations for USI A-3, A-4, and A-5 for the Steam Generator Tube Integrity Program. As a result, both the main steam line break (MSLB) and SGTR scenarios modeled here are expected to be significantly conservative even in the best estimate results given in Table 10.1.

As can be seen, all predicted core-melt frequencies are quite low (below 1E-06/py). The dominant sequences contributing to core-melt are the reactor coolant system overfill and overcool sequences. Three factors determined this dominance. First, the contribution to core-melt and risk from SGTR versus MSLB was dominated by the SGTR analysis regardless of the sequence examined. The overcool scenarios then had relatively high initiating frequency, and the sequences involved a steam line power operated relief valve (PORV) failure. This, then, constituted a main steam line failure, introducing directly the potential for a SGTR. The net result was a relatively high for core-melt and dose predicted for these sequences.

The probabilistic risk assessment (PRA) of the Zion plant was studied, where the conditional probability of core-melt given SGTR was 1E-05. However, steam line break due to water spillover could occur above the steam line MSIVs in this analysis. This type of failure significantly increases the potential for core damage given a subsequent SGTR, since isolation of the affected steam generator is not possible. The potential of exhausting injection water supplies before reactor coolant system (RCS) depressurization can be achieved is then raised significantly.

For those sequences involving a PORV or feedwater under operator control, the uncertainty in correct operator action has been identified in the analysis. Only Overfill Sequence 2 gives an unambiguous indication to the operator of changing steam generator levels. Thus the role of the operator in diagnosing and terminating the scenarios introduces some uncertainties. The analyses tried to treat these in a conservative fashion.

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TABLE 11.1. Summary of the INEL and PNL Estimates of Accident Initiator Frequencies, Core-Melt Frequencies and Public Risk

Sequence Initiator	INEL Accident Initiating Frequency Median (/py)	PNL Core-Melt Frequency Best Estimate (/py)	Public Risk Best Estimate (man-rem/py)
Steam Generator Overfill Sequence 1	1.4E-03		
Transient Shutdown		2.8E-09	1.5E-02
MSLB		7.7E-10	2.1E-03
SGTR		5.8E-08	2.8E-01
Subtotal		6.2E-08	3.0E-01
Steam Generator Overfill Sequence 2	5.4E-08		
Transient Shutdown		<1E-10	<1E-04
MSLB		<1E-10	<1E-04
SGTR		<1E-10	<1E-04
Subtotal		<1E-10	<1E-04
Reactor Coolant System Overcool Sequence 1	2.6E-07		
MSLB		<1E-10	<1E-04
SGTR		<1E-10	<1E-04
Subtotal		<1E-10	<1E-04
Reactor Coolant System Overcool Sequence 2	1.8E-02		
MSLB		1.1E-08	2.9E-02
SGTR		8.2E-07	3.9E+0
Subtotal		8.3E-07	3.9E+0
Reactor Coolant System Overpressure Sequence 1	1.5E-07	<1E-10	<1E-04
Reactor Coolant System Overpressure Sequence 2	3.7E-04	<1E-10	<1E-04
Steam Generator Tube Rupture Sequence 1	2.0E-03 7.0E-06 with SGTR	1.3E-08	6.9E-02
Steam Generator Tube Rupture Sequence 2	3.2E-03 1.1E-05 with SGTR	<1E-10	<1E-04
TOTAL		9.1E-07	4.3E+0

The role of the operator will also likely play an important role in reducing the frequency of control failures progressing to more serious accidents. This will likely introduce operator training or control room human factors engineering into any resolutions associated with this issue.

The overfill scenarios were assumed to lead to steam line break to provide any sort of core-melt sequence initiator. The basic uncertainty in the potential for inducing a steam line break still exists in the PWR analysis, as it did in the BWR analysis. Note, however, that in this case the overfill analyses did not actually progress to spillover of water into the steam lines. Power levels were also low or at startup for the PWR. The potential for water hammer and steam line break were adjusted accordingly compared to the BWR analysis, giving again what is thought to be a conservative assumption for steam line break.

The overpressure accident scenarios also had the potential for inducing a rupture in the reactor pressure vessel. However, the potential for vessel rupture due to pressurized thermal shock was determined to be low based on PTS program results from ORNL.

Finally, INEL also considered two SGTR events with aggravating control system failures, including loss of offsite power with PORV lift, and feedwater increase with PORV lift. The core-melt frequencies estimated were comparable to those for the overfill and overcool scenarios, but the combination of independent noncontrol-related failures to this core-melt frequency significantly reduced the role of any control modifications. Value/impact ratios were accordingly judged to be insignificant.

A total estimated frequency for core-melt of 9.1E-07/py and public risk of 4.3 man-rem/py are considered nondominant contributions to the overall core-melt frequency and risk of a PWR. For the WASH-1400 Surry Westinghouse PWR, the overall core-melt frequency was put at approximately 6E-05/py, with small break loss-of-coolant accidents (SBLOCAs) contributing approximately 2.9E-05/py. SGTRs were not modeled directly in the WASH-1400 PRA, but the system response would be similar to small LOCAs in which water did not return to sumps for recirculation. Because such LOCAs play a dominant role in the PWR risk, it is assumed that the risk calculated here will also represent only several percent of total overall plant risk.

11.1 PRESSURIZED THERMAL SHOCK

The overcool and overpressure scenarios postulated by INEL also have the potential for leading to pressurized thermal shock (PTS) events which could rupture the vessel and lead directly to core-melt. Overfill scenarios leading to MSLB or SGTR could likewise initiate a PTS event. The PTS program at ORNL is only now in the process of issuing a draft analysis of PTS events in the H. B. Robinson plant. However, an examination of the preliminary information available indicates that the potential for control failures leading to PTS and vessel rupture is very remote for the scenarios identified by INEL. Even with conservative estimates of the nil ductility transition temperature of 270°F

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for the H. B. Robinson plant, ORNL's modeling of a severe overcool due to steam line lift and HPI pressurization of the vessel appears to give vessel failure probabilities of 1E-04. However, the overall frequencies of core-melt were still estimated to be below those due to MSLB and SGTR. When the actual transition temperature of 130°F for the H. B. Robinson plant is used, the conditional probability of vessel rupture given PTS apparently drops to below 1E-06. The overall core-melt frequency predicted due to PTS thus becomes insignificant. The overpressure scenarios give insignificant core-melt frequencies.

The latter estimate is thought to be the best measure of risk due to PTS induced failures in the A-47 program. It is appropriate for conservative calculations to be used within the PTS program for evaluation and resolution of the PTS issue. However, these conservatisms should not be transferred into related issues if undue weighting or influence of specific safety concerns is to be avoided in a calculation of relative risk. This best estimate then indicates that PTS presently plays a minimal role in core-melt and risk for the A-47 analysis of this overcool scenario in the Westinghouse H. B. Robinson PWR.

11.2 VALUE/IMPACT ANALYSIS

An attempt was also made to propose several system modifications that might reduce the frequency of initiation of INEL-identified scenarios. These modifications (fixes) have been discussed in detail in Chapter 10. The public risk reductions and the cost associated with implementing each fix were also estimated, providing a value/impact ratio in terms of man-rem of public exposure reduced per $1000.

These results were presented in Table 10.1, with value/impact ratios ranging from approximately 0.3 man-rem/$1000 over a 30-year period to negligible values. The fixes that appear most promising for the PWR focus on an automatic high water level trip for the auxiliary feedwater (AFW), which reduces the potential for spillover by the AFW after a normal MFW trip, and automatic actuation of PORV block valves to protect against inadvertent PORV lifts. These modifications were estimated to have a value/impact ratio of 0.3 and 0.17 man-rem/$1000, respectively.

In the PWR, the overfill sequence driven by AFW overfill could be terminated by the simple addition of a high level AFW trip. The steam side break caused by PORV lift could likewise be eliminated through the use of block valves. Note that both of these fixes are already under operator control. Fixes to reduce the frequency of such events could focus on operator training and procedures, or on providing automatic actuation of shutoffs or block valves as assumed here. A more detailed examination of current procedures and time/signals available to the operator would be required to determine which modification is more appropriate.

As with the BWR, it is apparent that there is significant interaction between the operator and control systems. The role of the operator was conservatively estimated in the core-melt calculations presented above; however, no improvement in performance was postulated as a remedy to failures identified by INEL. With control of vessel water levels the focus of training and

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procedures upgrades since the TMI accident, a significant reduction in operator error for main or auxiliary feedwater failures, PORV lifts, etc., could reasonably be expected to reduce the progression of simple control failures to more serious accidents.

It is thus thought that the final resolution of A-47 should recognize the important role of the operator and make appropriate recommendations. Further consideration could include a more detailed examination of the time available to the operator, signals and indications available, and current procedures before requiring major equipment modifications. Recommendations could also be made to other specific task action items set up specifically to deal with operator actions during transients. These items are better geared to deal with the potential for reducing operator error in general, and would insure a consistent approach to operator interactions.

The addition of another level transmitter (LT) in a 2-out-of-4 trip logic does not appear to represent a cost-effective modification in the PWR to counteract the feedwater control system failures identified by INEL for feedwater overfill. The INEL estimate for trip failure with overfeed at 5.4E-08/yr makes such modifications pointless.

Note, however, that these conclusions are primarily dependent on the conservative assumptions used here for a high potential of steam line break and SGTR given the initiating control failures. If further analysis indicates that the potential for MSLB is much less or that isolation of the affected generator is possible (i.e., if breaks would occur preferentially below the MSIVs), then the resulting impact on core-melt frequency could easily be reduced by several orders of magnitude. Note also that the more prominent fixes such as for PORV lift are the subject of separate NRC safety action items.

The core-melt frequencies and public risk for all of the above scenarios are presented as best engineering estimates. The costs likewise represent best engineering estimate; thus a certain amount of latitude is needed in interpreting the value/impact ratio. However, the development of these accident initiators to core-melt is thought to reflect a conservative approach to estimating the impact of these failures on plant engineered safety systems. Cost estimates likewise tend to underestimate the true cost of nuclear plant modifications. These factors, when combined, tend to further reduce the estimated value/impact ratios given above.

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Pacific Northwest Laboratory (PNL) performed a probabilistic risk assessment to develop estimates of core-melt frequency and public risk due to control system failures in a Westinghouse pressurized water reactor. Value/impact analysis of proposed systems modifications to prevent the control system failures were also performed. Four control system failure modes were analyzed: 1) overfill, 2) overcool, 3) overpressure, and 4) steam generator tube rupture. For each mode, two failure sequences were postulated. These analyses were based on the results of failure modes and effects analyses previously performed at Idaho National Engineering Laboratory and conducted in support of the U.S. Nuclear Regulatory Commission's program for Unresolved Safety Issue A-47: Safety Implications of Control Systems.

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